U.S. Department of Energy
Office of River Protection
Contract Management Division
Mr. Michael K. Barrett
Contracting Officer
P.O. Box 450, MSIN H6-60
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CCN: 028980

Dear Mr. Barrett:

CONTRACT NO. DE-AC27-01RV14136 – TRANSMITTAL FOR INFORMATION: INCORPORATION OF SAFETY REQUIREMENTS DOCUMENT PAGE CHANGES

References:

1) CCN 027636, Letter, A. R. Veirup, BNI, to M. K. Barrett, ORP, "For Information, Incorporation of Safety Requirements Document and Integrated Safety Management Plan Page Changes Per U.S. Department of Energy Approval of Authorization Basis Change Notices ABCN-24590-01-00004, Revision 2, and 24590-WTP-ABCN-ESH-01-013, Revision 1," dated February 25, 2002

2) CCN 028213, Letter, R. C. Barr, OSR, to R. F. Naventi, BNI, "Office of Safety Regulation (OSR) Approval of Bechtel National, Inc (BNI) Authorization Basis Change Notice, 24590-WTP-ABCN-ESH-01-013, Revision 1, Codes and Standards Update/NPH Design Requirements," 02-OSR-0019, dated January 30, 2002

3) CCN 028170, Letter, R. C. Barr, OSR, to R. F. Naventi, BNI, "Office of Safety Regulation (OSR) Approval of Bechtel National, Inc (BNI)-Authorization Basis Change Notice ABCN-24590-01-00004, Revision 2, "Identification of Safety Analysis Report (SAR) Format and Content," 02-OSR-0034, dated January 29, 2002.

Authorization Basis Change Notice (ABCN) ABCN-24590-01-00004, Revision 2, was approved by the U.S. Department of Energy, Office of Safety Regulation (OSR) as noted in Reference 1. Authorization Basis Change Notice (ABCN) 24590-WTP-ABCN-ESH-01-013, Revision 1, was approved by the U.S. Department of Energy, Office of Safety Regulation (OSR) as noted in Reference 2. Page changes for the Safety Requirements Document (SRD), Revision 0b, were transmitted to the OSR to incorporate these ABCNs per Reference 1 and the OSR noted inconsistencies between the approved ABCNs and associated SRD page changes. The OSR requested revised page changes to SRD, Revision 0d.

Additional differences have been identified between the ABCN and SRD page changes and ABCN 24590-WTP-ABCN-ESH-01-013, Revision 1, is being revised to reflect these changes.

An electronic copy of revised page changes for SRD, Revision 0d, is also provided for the OSR's information and use.

Please contact Mr. Bill Spezialetti at (509) 371-4654 for any questions or comments on this transmittal.

Very truly yours,

A. R. Veirup
Prime Contract Manager

TR/slr

Attachment: *Safety Requirements Document*, 24590-WTP-SRD-ESH-01-001-02, Revision 0d

Document title:	**Safety Requirements Document Volume II**
Contract number:	DE-AC27-01RV14136
Department:	Environmental, Safety, and Health
Document number:	24590-WTP-SRD-ESH-01-001-02, Rev 0d
Date of issue:	6 March 2002
Issue status:	Approved

History Sheet

Rev	Date	Reason for revision	Revised by
0	21 Sep 2001	Supersedes BNFL-5193-SRD-01-02 Rev 5. Incorporates 24590-WTP-ABCN-ESH-01-017 Rev 0 changes (contractor-approved ABCN).	K Gibson
0a	4 Oct 2001	Incorporates ABCN-24590-01-00006 Rev 0 changes as partially approved by DOE Letter 01-OSR-0311 (CCN 023253) and contractor approved as revised ABCN-24590-01-00006, Rev 1.	K Gibson
0b	19 Feb 2002	Incorporates ABCN-24590-01-00004 Rev 2 changes as approved (with editorial changes specified) by DOE Letter 02-OSR-0034 (CCN 028170).	K Gibson
0c	20 Feb 2002	Incorporates 24590-WTP-ABCN-ESH-01-013 Rev 1 as approved by DOE Letter 02-OSR-0019 (CCN 028213).	K Gibson
0d	6 Mar 2002	This revision makes minor corrections to changes incorporated at Revisions 0b and 0c.	K Gibson

Revision Status

Page No.	Rev.	Page No.	Rev.	Page No.	Rev.	Page No.	Rev.
i	0d	4-15	0	7-4	0	9-11	0a
ii	0d	4-16	0a	7-5	0	9-12	0a
iii	0d	4-17	0	7-6	0a	9-13	0a
iv	0d	4-18	0	7-7	0a	9-14	0a
v	0a	4-19	0	7-8	0a	A-i	0
vi	0b	4-20	0	7-9	0	A-ii	0
1-1	0a	4-21	0	7-10	0	A-1	0
1-2	0	4-22	0	7-11	0a	A-2	0
1-3	0a	4-23	0	7-12	0a	A-3	0
1-4	0	4-24	0	7-13	0	A-4	0
2-1	0	4-25	0	7-14	0a	A-5	0a
2-2	0	4-26	0	7-15	0	A-6	0
2-3	0a	4-27	0	7-16	0	A-7	0
3-1	0a	4-28	0	7-17	0a	A-8	0
3-2	0a	4-29	0	7-18	0	A-9	0
3-3	0a	4-30	0	7-19	0a	A-10	0
3-4	0a	4-31	0	7-20	0a	A-11	0a
3-5	0a	4-32	0a	7-21	0a	A-12	0
3-6	0	5-1	0	7-22	0	A-13	0
3-7	0	5-2	0	7-23	0	A-14	0
3-8	0	5-3	0	7-24	0	A-15	0
3-9	0	5-4	0	7-25	0a	A-16	0
4-1	0a	5-5	0	7-26	0a	A-17	0
4-2	0	5-6	0	7-27	0a	A-18	0
4-3	0	5-7	0	8-1	0	A-19	0
4-4	0d	5-8	0	9-1	0	A-20	0
4-5	0c	5-9	0	9-2	0	B-i	0
4-6	0	5-10	0	9-3A	0b	B-ii	0
4-7	0	5-11	0	9-3B	0b	B-1	0
4-8	0d	5-12	0	9-4	0d	B-2	0
4-9	0d	5-13	0	9-5	0a	B-3	0
4-10	0d	6-1	0a	9-6	0	B-4	0
4-11	0	6-2	0a	9-7	0	B-5	0
4-12	0	7-1	0	9-8	0	B-6	0
4-13	0	7-2	0	9-9	0	B-7	0
4-14	0	7-3	0a	9-10	0a	B-8	0

Page No.	Rev.	Page No.	Rev.	Page No.	Rev.	Page No.	Rev.
B-9	0	C-4	0	D-4	0	G-5	0b
B-10	0	C-5	0	D-5	0	G-6	0b
B-11	0	C-6	0	D-6	0	G-7	0b
B-12	0	C-7	0	D-7	0	G-8	0b
B-13	0	C-8	0	D-8	0	G-9	0b
B-14	0	C-9	0	D-9	0	G-10	0b
B-15	0	C-10	0	D-10	0	G-11	0b
B-16	0	C-11	0	D-11	0	G-12	0b
B-17	0	C-12	0c	D-12	0	G-13	0b
B-18	0	C-13	0	E-i	0	G-14	0b
B-19	0	C-14	0	E-1	0	G-15	0b
B-20	0	C-15	0	E-2	0	G-16	0b
B-21	0	C-16	0	F-i	0	G-17	0b
B-22	0	C-17	0c	F-ii	0	G-18	0b
B-23	0	C-18	0c	F-1	0	G-19	0b
B-24	0	C-19	0c	F-2	0	G-20	0b
B-25	0	C-20	0c	G-i	0b	G-21	0b
C-i	0	D-i	0	G-ii	0b	G-22	0b
C-ii	0	D-ii	0	G-1	0b	G-23	0b
C-1	0	D-1	0	G-2	0b	G-24	0d
C-2	0	D-2	0	G-3	0b	G-25	0b
C-3	0	D-3	0	G-4	0b		

Approved by: ______________________ ______________________ Date: ______________
(Print name) (Sign)

CONTENTS

APPENDICES

1.0 Radiological, Nuclear and Process Safety Objectives

Safety Criterion: 1.0 - 1

A comprehensive radiological and process safety management program shall be used to eliminate or reduce the incidence, or mitigate the consequences of, accidental radioactive or chemical releases, process fires, and process explosions. This program shall address management practices, technologies, and procedures. Radiological and process safety management shall confirm that the facility is properly designed, the integrity of the design is maintained, and the facility is operated according to the safe manner intended.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 4.1 Safety Management Processes
Chapter: 5.0 Process Safety Management

Regulatory Basis

DOE/RL-96-0006 5.1.1 Process Safety Management
DOE/RL-96-0006 5.1.2 Process Safety Objective

Safety Criterion: 1.0 - 2

Principal emphasis shall be placed on the prevention of accidents, particularly any that could cause an unacceptable release, as the primary means of achieving safety.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification
DOE IG Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria, 2.3
DOE Order 420.1 Facility Safety, 4.1.1.2

Regulatory Basis

DOE/RL-96-0006 4.1.1.2 Defense in Depth-Prevention

Safety Criterion: 1.0 - 3

The risk, to an average individual within 1 mile of the RPP-WTP Controlled Area Boundary, of prompt fatalities that might result from an accident shall not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents which members of the U.S. population generally are exposed.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification
24590-WTP-SRD-ESH-01-001-02, Appendix D, Radiological Exposure Standards for the RPP-WTP Project

Regulatory Basis

DOE/RL-96-0006 3.1.2 Accident Risk Goal

Safety Criterion: 1.0 - 4

The risk, to the public and workers within 16 km (10 miles) of the RPP-WTP, of cancer fatalities that might result from RPP-WTP operations shall not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks to which members of the U.S. population generally are exposed.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

24590-WTP-SRD-ESH-01-001-02, Appendix D, Radiological Exposure Standards for the RPP-WTP Project

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Regulatory Basis

DOE/RL-96-0006 *3.1.1* *Operations Risk Goal*

Safety Criterion: 1.0 - 5

The risk to workers within the RPP-WTP Controlled Area Boundary, of fatality from radiological exposure that might result from an accident, shall not be a significant contributor to the overall occupational risk of fatality to workers.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

24590-WTP-SRD-ESH-01-001-02, Appendix D, Radiological Exposure Standards for the RPP-WTP Project

Regulatory Basis

DOE/RL-96-0006 *3.1.3* *Worker Accident Risk Goal*

Safety Criterion: 1.0 - 6

Measures in the design and operation of the facility to protect the public, workers, and environment against accident conditions shall be evaluated using an acceptable approach to demonstrate that they perform their intended purpose with high confidence.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix B, Implementing Standard for Defense in Depth

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis

DOE/RL-96-0006 *3.3.1* *Public Protection*

DOE/RL-96-0006 *3.3.2* *Worker Protection*

Safety Criterion: 1.0 - 7

To compensate for potential human and equipment failures, a defense-in-depth strategy shall be applied to the facility commensurate with the hazards; such that, as appropriate to control the risk, safety is vested in multiple, independent safety provisions, no one of which is to be relied upon excessively to protect the public, the workers, or the environment. This strategy shall be applied to the design and operation of the facility.

Implementing Codes and Standards

ANSI/ANS 58.9-1981 Single Failure Criteria for Light Waster Reactor Safety-Related Fluid Systems
24590-WTP-SRD-ESH-01-001-02, Appendix B, Implementing Standard for Defense in Depth
DOE IG Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria, 2.3
DOE Order 420.1 Facility Safety 4.1.1.2
IEEE 379-1994 Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems

Regulatory Basis

DOE/RL-96-0006 *4.1.1.1 Defense in Depth-Defense in Depth*

Safety Criterion: 1.0 - 8

Structures, systems, and components (SSCs) that serve to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the workers and the public are classified as Important to Safety. It encompasses the broad class of facility features addressed (not necessarily explicitly) in the top-level radiological, nuclear, and process safety standards and principles that contribute to the safe operation and protection of workers and the public during all phases and aspects of facility operations (i.e., normal operation as well as accident mitigation). This definition includes not only those structures, systems, and components that perform safety functions and traditionally have been classified as safety class, safety-related or safety-grade, but also those that place frequent demands on or adversely affect the performance of safety functions if they fail or malfunction, i.e., support systems, subsystems, or components. Thus, these latter structures, systems, and components would be subject to applicable top-level radiological, nuclear, and process safety standards and principles to a degree commensurate with their contribution to risk. In applying this definition, it is recognized that during the early stages of the design effort all significant systems interactions may not be identified and only the traditional interpretation of Important to Safety, i.e., safety-related may be practical. However, as the design matures and results from risk assessments identify vulnerabilities resulting from non-safety-related equipment, additional structures, systems, and components should be considered for inclusion within this definition.

Important to Safety includes SSCs designated as Safety Design Class and Safety Design Significant.

Safety Design Class SSCs includes those that, by performing their specified safety function, prevent workers or the maximally exposed member of the public from receiving a radiological or chemical exposure that exceeds the exposure standards defined in the SRD. Those features credited for the prevention of a criticality event are also designated as Safety Design Class.

Safety Design Significant SSCs are those needed to achieve compliance with the radiological or chemical exposure standards for the public and workers during normal operation; and SSCs that can, if they fail or malfunction, place frequent demands on, or adversely affect the function of, Safety Design Class SSCs.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

24590-WTP-SRD-ESH-01-001-02, Appendix D, Radiological Exposure Standards for the RPP-WTP Project

Regulatory Basis

DOE/RL-96-0006 3.3.1 Public Protection
DOE/RL-96-0006 3.3.2 Worker Protection

Safety Criterion: 1.0 - 9

The RPP-WTP Contractor shall accept responsibility for the safety of the RPP-WTP.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Chapter: 1.0 Project Safety Approach
Section: 11.1 Design and Construction Phase
Section: 11.2 Operations Phase

Regulatory Basis

DOE/RL-96-0006 4.1.2.1 Safety Responsibility-Safety Responsibility
DOE/RL-96-0006 4.3.1.1 Conduct of Operations-Organizational Structure
DOE/RL-96-0006 5.1.3 Process Safety Responsibility

Safety Criterion: 1.0 - 10

In addition to the Safety Criteria contained herein, compliance with all requirements of 10 CFR 830.120 and 10 CFR 835 shall be achieved absent the granting of an exemption request to any specific requirement therein.

Regulatory Basis

10 CFR 830.120 Quality assurance requirements Location
10 CFR 835 Occupational Radiation Protection Location: 1
DE-AC06-96RL13308 Part I Section C.5 Table S4-1

2.0 Radiological and Process Standards

Safety Criterion: 2.0 - 1

The following Radiological Dose Standards shall be applied to protect the public and workers from RPP-WTP radiological hazards.

Table 2-1. Radiological Exposure Standards Above Normal Background (Sheet 1)

Description	Estimated Frequency of Occurrence f (yr^{-1})	General Guidelines	Worker	Collocated Worker	Public
Normal Events Events that occur regularly in the course of facility operation (e.g., normal facility operations); including routine and preventive maintenance activities.	>0.1	Normal modes of operating facility systems should provide adequate protection of health and safety.	≤5 rem/yr ≤50 rem/yr any organ, skin, or extremity ≤15 rem/yr lens of eye ≤1.0 rem/yr ALARA design objective per 10CFR835.1002(b)[(1)]	≤5 rem/yr ≤1.0 rem/yr ALARA design objective per 10 CFR 835.1002(b)[(1)]	≤10 mrem/yr (airborne pathway) ≤100 mrem/yr (all sources) ≤100 mrem/yr (public in the controlled area) ≤25 mrem/yr (radioactive waste)
Anticipated Events Events of moderate frequency that may occur once or more during the life of a facility (e.g., minor incidents and upsets).	$10^{-2}<f\leq10^{-1}$	The facility should be capable of returning to operation without extensive corrective action or repair.	≤5 rem/event [(2,3)] 1.0 rem/event design action threshold [(4)]	≤5 rem/event (2, 3) 1.0 rem/event design action threshold[(4)]	≤100 mrem/event[(3)]
Unlikely Events Events that are not expected, but may occur during the lifetime of a facility (e.g., more severe incidents).	$10^{-4}<f\leq10^{-2}$	The facility should be capable of returning to operation following potentially extensive corrective action or repair, as necessary.	≤25 rem/event[(2,3)]	≤25 rem/event[(2, 3)]	≤5 rem/event[(3)]

Table 2-1. Radiological Exposure Standards Above Normal Background (Sheet 2)

Description	Estimated Frequency of Occurrence f (yr^{-1})	General Guidelines	Worker	Collocated Worker	Public
Extremely Unlikely Events Events that are not expected to occur during the life of the facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material.	$10^{-6}<f\leq10^{-4}$	Facility damage may preclude returning to operation.	≤25 rem/event[(2,3)]	≤25 rem/event[(2,3)]	≤25 rem/event ≤5 rem/event target[(3)] ≤300 rem/event to thyroid
Location of Receptor			Within the RPP-WTP Controlled Area Boundary	The most limiting location at or beyond the RPP-WTP Controlled Area Boundary	The most limiting location along the near river bank/ Hwy240/ southern boundary

Notes (1) In addition to meeting the listed design objective of 10 CFR 835.1002(b), the inhalation of radioactive material by workers and collocated workers under normal conditions is kept ALARA through the control of airborne radioactivity as described in 10 CFR 835.1002(c).

(2) In addition to meeting the listed worker and collocated worker exposure standards for accidents, the Worker Accident Risk Goal is satisfied through the calculation of the risk from accidents with accident prevention and mitigation features added as necessary to meet the Goal.

(3) In addition to meeting the listed exposure standards for accidents, the approach to accident mitigation is to evaluate accident consequences to ensure that the calculated exposures are far enough below standards to account for uncertainties in the analysis, and to provide for sufficient design margin and operational flexibility.

(4) When a calculated accident exposure exceeds this threshold, then appropriate actions are taken. These include carrying out a less bounding (i.e., more realistic) evaluation to show that the accident consequences will be below the threshold or evaluating additional safeguards for cost-effectiveness and/or feasibility. This threshold is not a limit; it does not require the implementation of additional preventative or mitigative features if they are not both cost-effective and feasible.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

24590-WTP-SRD-ESH-01-001-02, Appendix D, Radiological Exposure Standards for the RPP-WTP Project

Regulatory Basis

DOE/RL-96-0006 2.1 Individual (Dose Standards Above Normal Background)

Safety Criterion: 2.0 - 2

The following dose standards shall be applied to protect the public and workers from RPP-WTP chemical hazards.

Releases exposing the offsite public to ERPG-2 concentrations (AIHA 1999)

Releases exposing the co-located worker to ERPG-3 concentrations (AIHA 1999)

Accidents affecting the facility worker that could cause in-patient hospitalization of at least 3 facility workers, or at least a single fatality.

Where ERPG values have not been published, the DOE Temporary Emergency Exposure Limits (TEELs) may be used as substitute ERPGs.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Safety Criterion: 2.0 - 3

In addition to the dose limits specified for the public in Safety Criterion 2.0-1 Table 2-1, the dose in any unrestricted area from external sources shall not exceed 0.002 rem in any one hour.

Implementing Codes and Standards

DOE G 441.1-2, Occupational ALARA Program Guide

Regulatory Basis

WAC 246-221 Radiation Protection Standards Location: 060 (1)
WAC 246-247 Radiation Protection - Air Emissions Location: Part 040 (2)

3.0 Nuclear and Process Safety

3.1 Hazards Analysis

Safety Criterion: 3.1 - 1

An initial process hazard analysis (hazard evaluation) shall be performed using acceptable industry practices. The analysis shall include consideration of both chemical and radiological hazards. The process hazard analysis shall be appropriate to the complexity of the process and shall identify, evaluate, and document the design features which control the hazards involved in the process.

The process hazard analysis shall be performed by a team with expertise in engineering and process operations, and the team shall include at least one member who has experience and knowledge specific to the process being evaluated. Also, one member of the team must be knowledgeable in the specific process hazard analysis methodology being used.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document Volume II
Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis

DOE/RL-96-0006 *5.2.2* *Process Hazard Analysis*

Safety Criterion: 3.1 - 2

A compilation of written process safety information appropriate to the stage of design being considered shall be completed to support the process hazard analysis. The compilation of written process safety information enables the employer and the employees involved in operating the process to identify and understand the hazards posed by those processes involving radioactive materials and process chemicals considered to pose a hazard. This process safety information shall include information pertaining to the hazards of the materials used or produced by the process, information pertaining to the technology of the process, and information pertaining to the equipment in the process.

(1) Information pertaining to the hazards of the materials in the process including:

(a) Toxicity information

(b) Permissible exposure limits

(c) Physical data

(d) Reactivity data

(e) Corrosivity data

(f) Thermal and chemical stability data

(g) Hazardous effects of inadvertent mixing of different materials that could foreseeably occur

(2) Information pertaining to the technology of the process including at least the following:
   (a) A block flow diagram or simplified process flow diagram
   (b) Process chemistry
   (c) Maximum intended inventory
   (d) Safe upper and lower limits for such items as temperatures, pressures, flows or compositions
   (e) An evaluation of the consequences of deviations, including those affecting the safety and health of employees

(3) Information pertaining to the equipment in the process including:
   (a) Materials of construction
   (b) Process drawings or piping and instrument diagrams (P&IDs)
   (c) Electrical classification
   (d) Relief system design and design basis
   (e) Ventilation system design
   (f) Design codes and standards employed
   (g) Material and energy balances
   (h) Safety systems (e.g. interlocks, detection or suppression systems)

The records shall be maintained documenting that equipment complies with recognized and generally accepted good engineering practices. The safety information shall be kept up-to-date.

Implementing Codes and Standards

Regulatory Basis

DOE/RL-96-0006	*5.2.1*	*Process Safety Information*
DOE/RL-96-0006	*5.2.2*	*Process Hazard Analysis*

Safety Criterion: 3.1 - 3

The process hazard analysis shall address:

(1) The hazards of the process

(2) Engineering and administrative controls applicable to the hazards and their interrelationships such as appropriate application of detection methodologies to provide early warning of releases. (Acceptable detection methods might include process monitoring and control instrumentation with alarms, and detection hardware.)

(3) Consequences of failure of engineering and administrative controls

(4) Facility siting

(5) Human factors

(6) A qualitative evaluation of a range of the possible safety and health effects of failure of controls on employees in the workplace

(7) Common-mode and common-cause failure events

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document Volume II
Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis

DOE/RL-96-0006 *5.2.2* *Process Hazard Analysis*

Safety Criterion: 3.1 - 4

The hazard analysis shall be performed in accordance with the following requirements:

(1) The consequences of unmitigated releases of radioactive material and process chemicals considered to pose a hazard shall be evaluated.

(2) The hazard analysis shall be based on an inventory of all radioactive and hazardous nonradioactive materials that are stored, utilized, or may be formed within the facility.

(3) The hazard analysis shall identify energy sources or processes that might contribute to the generation or uncontrolled release of radioactive or process chemicals considered to pose a hazard. The hazard analysis shall estimate the consequences of accidents in which the facility or process and/or materials in the inventory are assumed to interact, react, or be released in a manner to produce a threat or challenge to the health and safety of individuals on-site and off site.

(4) The risks that hazardous inventories and energy sources present shall be evaluated by consideration of normal operation (including startup, testing, and maintenance), anticipated operational occurrences, and accident conditions. The identification of anticipated operational occurrences and accident conditions shall consider internal events (i.e., equipment failure and human error), external events (e.g., nearby facilities and transportation), and natural phenomena.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document Volume II
Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis

DOE/RL-96-0006	*3.3.3*	*Accident Vulnerability Mitigation*
DOE/RL-96-0006	*5.2.2*	*Process Hazard Analysis*

Safety Criterion: 3.1 - 5

A written plan of action shall be developed regarding employee participation in the conduct and development of process hazards analyses and on the development of process safety management. Employees and their representatives shall be consulted on the conduct and development of process hazards analyses and on the development of the other elements of process safety management. Employees and their representatives shall be provided access to process hazard analyses and other information developed related to process safety.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document Volume II
Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis

Safety Criterion: 3.1 - 6

A system shall be established to promptly address the hazard analysis team's findings and recommendations; assure that the recommendations are resolved in a timely manner; and that the resolution is documented. The contractor shall document what actions are to be taken; complete actions; develop a written schedule of when these actions are to be completed; communicate the actions to operating, maintenance and other employees whose work assignments are in the process and who may be affected by the recommendations or actions.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02
Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis

DOE/RL-96-0006 5.2.2 Process Hazard Analysis

Safety Criterion: 3.1 - 7

The process hazard analysis shall be updated to reflect changes concurrently with the annual update of the FSAR by a qualified team, to assure that the process hazard analysis is consistent with the current process.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.3.3 Changes to Safety Documentation
Section: 5.6.2 Updating of the Hazard Analysis Report

Regulatory Basis

DOE/RL-96-0006 5.2.2 Process Hazard Analysis

Safety Criterion: 3.1 - 8

Employers shall retain process hazards analyses and updates or revalidations as well as the documented resolution of any recommendations for the life of the process.

Implementing Codes and Standards

Regulatory Basis

3.2 Accident Analysis

Safety Criterion: 3.2 - 1

Acceptable risk analyses shall be applied during the design to delineate provisions for the prevention and mitigation, including emergency preparedness and response, of otherwise risk-dominant events.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3 Description of the Integrated Safety Management Plan
Section: 3.10 Emergency Preparedness

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis

DOE/RL-96-0006 *4.2.1.2 Design-Risk Assessment*

3.3 Criticality

Safety Criterion: 3.3 - 1

The facility shall be designed and operated in a manner that prevents nuclear criticality.

Implementing Codes and Standards

ANSI/ANS 8.1-1983 (R 1988) Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors

ANSI/ANS 8.19-1996 Administrative Practices for Nuclear Criticality Safety

Regulatory Basis

DOE/RL-96-0006 4.2.2.5 Proven Engineering Practices/Margins-Criticality

Safety Criterion: 3.3 - 2

The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and in the nature of the immediate environment under accident conditions.

The multiplication factor (k<eff>), including all biases and uncertainties at a 95% confidence level, shall be shown to not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Safety Criterion: 3.3 - 3

Process designs shall incorporate sufficient factors of safety to require at least two unlikely, independent, and concurrent changes in process conditions before a criticality accident is possible. Protection shall be provided by either:

(1) the control of two independent process parameters (which is the preferred approach, when practical, to prevent common-mode failure), or

(2) a system of multiple controls on a single process parameter.

The number of controls required for a single controlled process parameter shall be based upon control reliability and any features that mitigate the consequences of control failure. In all cases, no single credible event or failure shall result in the potential for a criticality accident.

An exception to the application of double contingency, where single contingency operations are permissible, is presented in paragraph 5.1 of ANSI/ANS-8.10-1983, R88. This exception applies to operations with shielding and confinement (e.g., hot cells or other shielded facilities).

Double contingency shall be demonstrated by documented evaluations.

Safety Criterion: 3.3 - 4

Where a sufficient quantity of fissionable material is being processed such that criticality safety is a concern, passive engineered controls, such as geometry control, shall be considered as the preferred control method. Where passive engineered control is not feasible, the preferred order of controls is active engineered controls followed by administrative controls. The double contingency analysis shall justify the chosen controls. Full advantage may be taken of any nuclear characteristics of the process materials and equipment. The geometry must be considered as water moderated and reflected unless it can be shown the presence of water is not credible. All dimensions, nuclear properties, and other features upon which reliance is placed shall be documented and verified prior to beginning operations, and control shall be exercised to maintain them.

Safety Criterion: 3.3 - 5

To protect against an uncontrolled nuclear criticality incident, nuclear criticality safety considerations and controls shall be evaluated for accidents, normal operations, and before any significant operational changes are made.

Safety Criterion: 3.3 - 6

Criticality Accident Alarm Systems (CAS) and Criticality Detection Systems (CDS) shall be required as follows:

(1) In those locations where the mass of fissionable material exceeds the limits established in Table 3-1 Inventory of Fissionable Material and the probability of a criticality accident is greater than 1E-06 per year, a CAS conforming to ANSI/ANS-8.3-1986 shall be provided to cover occupied areas in which the expected dose exceeds 12 rads (0.12 greys) in free air, where a CAS is defined to include a criticality accident detection device and a personnel evacuation alarm.

(2) In those locations where the mass of fissionable material exceeds the limits established in Table 3-1 Inventory of Fissionable Material and the probability of a criticality accident is greater than 1E-06 per year, but there are no occupied areas in which the expected dose exceeds 12 rads (0.12 greys) in free air, a CDS shall be provided, where a CDS is defined to be an appropriate criticality accident detection device but without an immediate evacuation alarm. The CDS response time should be sufficient to allow for appropriate process-related mitigation and recovery actions. Appropriate response guidance to minimize personnel exposure shall be provided.

(3) In those locations where the mass of fissionable material exceeds the limits established in Table 3-1 Inventory of Fissionable Material, but a criticality accident is determined to be impossible due to the physical form of the fissionable material, or the probability of occurrence is determined to be less than 1E-06 per year, neither a CAS nor a CDS is required. Neither a CAS nor a CDS is required for fissionable material during shipment when packaged in approved shipping containers, or when packaged in approved shipping containers awaiting transport provided that no other operation involving fissionable material not so packaged is permitted on the shipping dock or in the shipment area.

(4) If a criticality accident is possible wherein a slow (i.e., quasistatic) increase in reactivity could occur leading from subcriticality to criticality to self-shutdown without initiating emplaced criticality alarms, CASs should be supplemented by warning devices such as audible personnel dosimeters (e.g., pocket chirpers/flashers, or their equivalents), area radiation monitors, area dosimeters, or integrating CASs to aid in protecting workers against the consequences of slow criticality accidents.

(5) Neither a CAS nor a CDS is required to be installed for handling or storage of fissionable material when sufficient shielding exists that is adequate to protect personnel (e.g., hot cells); however, a means to detect fission product gases or other volatile fission products shall be provided in occupied areas immediately adjacent to such shielded areas, except for systems where no fission products are likely to be released.

Note: The frequency of 1E-06 per year is used as a measure of credibility and does not require a probabilistic risk assessment be performed. Reasonable grounds for incredibility may be presented on the basis of commonly accepted engineering judgement.

Table 3-1. Inventory of Fissionable Material[1]

Isotope	Inventory in Individual Unrelated Area
U-235	700g
U-233	520g
Pu-239	450g
Any Combination of above Isotopes	450g

[1] Per ANSI/ANS-8.3-1986 paragraph 4.2.1

Safety Criterion: 3.3 - 7

The monitoring system shall be capable of detecting a criticality that produces an absorbed dose in soft tissue of 20 rads (0.20 greys) of combined neutron and gamma radiation at an unshielded distance of 2 meters from the reacting material within one minute.

Safety Criterion: 3.3 - 8

Coverage of all areas requiring detection may be provided by a single detector.

4.0 Engineering and Design

Safety Criterion: 4.0 - 1

Formal configuration management shall be applied to all facility activities through deactivation of the RPP-WTP to ensure that programmatic objectives, including safety, are fully achieved. Work shall be performed and controlled according to pre-approved plans and procedures that clearly delineate responsibility. Documented records shall be retained.

Implementing Codes and Standards

ISO 10007 Quality Management - Guidelines for Configuration Management

Regulatory Basis

DOE/RL-96-0006 *4.1.5.1* *Configuration Management-Formal Configuration Management*

Safety Criterion: 4.0 - 2

Written procedures shall be established and implemented to manage changes (except for "replacements in kind") to process chemicals, technology, equipment, and procedures; and, changes to facilities that affect a covered process. The procedures shall assure that the following considerations are addressed prior to any change:

(1) The technical basis for the proposed change

(2) Impact of change on safety and health

(3) Modifications to operating procedures

(4) Necessary time period for the change

(5) Authorization requirements for the proposed change

Employees involved in operating a process and maintenance and subcontract employees whose job tasks will be affected by a change in the process shall be informed of, and trained in, the change prior to start-up of the process or affected part of the process. If a change covered by this paragraph results in a change in the process safety information, such information shall be updated accordingly. If a change covered by this paragraph results in a change in operating procedures or practices, such procedures or practices shall be updated accordingly.

Implementing Codes and Standards

ISO 10007 Quality Management - Guidelines for Configuration Management

Regulatory Basis

DOE/RL-96-0006 *5.2.9* *Management of Change*

Safety Criterion: 4.0 - 3

A system shall be used to control and maintain accurate as-built records for Important to Safety SSCs through deactivation of the facility.

Implementing Codes and Standards

ISO 10007 Quality Management - Guidelines for Configuration Management

Regulatory Basis

DOE/RL-96-0006 *4.1.5.3 Configuration Management-Design Documentation*

4.1 General Design

Safety Criterion: 4.1 - 1

The facility design shall provide for the prevention and mitigation of the risks associated with radiological and chemical material inventories and energy sources. The facility design shall include consideration of normal operation (including startup, testing and maintenance), anticipated operational occurrences, external events, and accident conditions.

Prevention shall be the preferred means of achieving safety.

Defense-in-depth shall be applied commensurate with the hazard to provide multiple physical and administrative barriers against undue radiation and chemical exposure to the public and workers.

Implementing Codes and Standards

ANSI/ANS 58.9-1981 Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems
24590-WTP-SRD-ESH-01-001-02, Appendix B, Implementing Standard for Defense in Depth
DOE IG Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria, 2.3
DOE Order 420.1 Facility Safety 4.1.1.2
IEEE 379-1994 Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems

Regulatory Basis

DOE/RL-96-0006	*4.1.1.1*	*Defense in Depth-Defense in Depth*
DOE/RL-96-0006	*4.1.1.2*	*Defense in Depth-Prevention*
DOE/RL-96-0006	*4.2.1.1*	*Design-Safety Design*

Safety Criterion: 4.1 - 2

Structures, systems, and components designated as Important to Safety shall be designed, fabricated, erected, constructed, tested, inspected, and maintained to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components designated as Important to Safety shall be maintained through deactivation of the facility.

Items and processes shall be designed using sound engineering/scientific principles and appropriate standards.

Design features that enhance the margin of safety through simplified, inherently safe, passive, or other highly reliable means to accomplish the specified safety function should be employed to the maximum extent practical.

Design work, including changes, shall incorporate applicable requirements and design bases. Design interfaces shall be identified and controlled. The adequacy of design products shall be verified or validated by individuals or groups other than those who performed the work. Verification and validation work shall be completed before approval and implementation of the design.

Safety technologies incorporated into the facility design should have been proven by experience or testing and should be reflected in approved codes and standards. Significant new design features should be introduced only after thorough research and model or prototype testing at the component, system, or facility level, as appropriate, to achieve the necessary level of confidence that the design feature will perform as expected.

Implementing Codes and Standards

ACI 318-99, *Building Code Requirements for Structural Concrete*
ACI 318R-99, *Commentary on Building Code Requirements for Structural Concrete*
ACI 349-01, *Code Requirements for Nuclear Safety-Related Concrete Structures*
ACI 349R-01, *Commentary on Code Requirements for Nuclear Safety-Related Concrete Structures*
AISC MO16-89, *Manual for Steel Construction - Allowable Stress Design, Ninth Edition*
ANSI/AISC N690-94, *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities*
ASCE 4-98, *Seismic Analysis of Safety-Related Nuclear Structures and Commentary*
ASCE 7-95, *Minimum Design Loads for Buildings and Other Structures*
DOE-STD 1020-94 (Change 1, 1996), *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*
1997, *UBC Uniform Building Code*
DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98
ACI 530-99, *Building Code Requirements for Masonry Structures and Commentary*
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
ISO 10007, *Quality Management - Guidelines for Configuration Management*
ASTM D3740, *Standard Practice for Minimum Requirements for Agencies Engaged in the Testing and/or Inspection of Soil and Rock as Used in Engineering Design and Construction*
ASTM D2922, *Standard Test Method for Laboratory Determination of Moisture Content of Soil*
ASTM D3017, *Standard Test Method for Water Content of Soil and Rock in Place by Nuclear Methods*
PCA Publication, EB 080.01, *Strength Design of Anchorage to Concrete*

Regulatory Basis

DOE/RL-96-0006	*4.1.2.4*	*Safety Responsibility-Operating Experience and Safety Research*
DOE/RL-96-0006	*4.1.5.1*	*Configuration Management-Formal Configuration Management*
DOE/RL-96-0006	*4.1.6.2*	*Quality Assurance-Established Techniques and Procedures*
DOE/RL-96-0006	*4.2.2.1*	*Proven Engineering Practices/Margins-Proven Engineering Practices*
DOE/RL-96-0006	*4.2.2.3*	*Proven Engineering Practices/Margins-Safety System Design and Qualification*
DOE/RL-96-0006	*4.2.5.1*	*Inherent/Passive Safety Characteristics-Safety Margin Enhancement*

Safety Criterion: 4.1 - 3

This criterion addresses natural phenomena hazards (NPH) design for structures, systems, and components (SSCs) that are Important to Safety and have NPH safety functions.

SSCs designated as Important to Safety (i.e., Safety Design Class and Safety Design Significant) shall be designed to withstand the effects of NPH events such as earthquakes, wind, and floods without loss of capability to perform specified safety functions required as the result of the NPH events. This includes both the front line and support systems that must function for a NPH event such that the public, collocated worker, or facility worker exposure standards of Safety Criterion 2.0-1 or 2.0-2 are not exceeded.

SSCs that are designated Safety Design Class (excepting those so designated based solely on chemical hazards) and that are required to perform a safety function as a result of a given NPH shall be designed to withstand the NPH loadings of that NPH as provided in Table 4-1. These SSCs are designated Seismic Category I (SC-I) for earthquakes and Performance Category 3 (PC-3) for other NPH.

SSCs that are designated Safety Design Significant (excepting those so designated based solely on chemical hazards) whose continued function is not required for an NPH event, but whose failure as a result of an NPH event could reduce the functioning of a Safety Design Class SSC such that exposure standards might be exceeded, shall be designed to withstand the NPH loadings of that NPH as provided in Table 4-1. For these SSCs, however, for seismic response only, credit may be taken for inelastic energy absorption per Table 2-4 of DOE-STD-1020-94. These SSCs are designated SC-II for earthquakes and PC-3 for other NPH.

For any SSC included under this criterion, other NPH loads (for which the SSC has no safety function) may be taken from Safety Criterion 4.1-4 and Table 4-2 in lieu of Safety Criterion 4.1-3 and Table 4-1.

Table 4-1 Natural Phenomena Design Loads for Important to Safety SSCs with NPH Safety Functions

Hazard	Load	Source Document for Load
Seismic	DBE with 0.26 *g* horizontal PGA and 0.18 *g* vertical PGA See Figures 4-1 and 4-2	WHC-SD-W236A-TI-002 [a] DOE-STD-1020-94[b]
Straight wind	111 mi/hr , 3-second gust, at 33 ft above ground, Importance factor, I=1.0	DOE Newsletter [c]
Wind Missile	2x4 timber plank, 15 lb at 50 mi/hr (horiz), Max height 30 ft	DOE-STD-1020-94 [b]
Tornado and Tornado Missiles	Not Applicable	DOE-STD-1020-94 [b]
Volcanic ash	12.5 lb/ft^2	HNF-SD-GN-ER-501 [d]
Flooding	Dry site for river flooding Local precipitation: 4 in. for 6 hours	HNF-SD-GN-ER-501 [d]
Snow	15.0 lb/ft^2 snow load	HNF-SD-GN-ER-501 [d]

[a] Geomatrix, 1996, *Probabilistic Seismic Hazard Analysis DOE Hanford Site, Washington,* WHC-SD-W236A-TI-002, Rev.1A, prepared for Westinghouse Hanford Company, Richland, Washington.

[b] DOE STD-1020-94, (1996, Change 1) *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, U.S. Department of Energy, Washington, D.C., 1996.

[c] DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98.

[d] HNF-SD-GN-ER-501, Rev. 1, "Natural Phenomena Hazards, Hanford Site, South-Central Washington", Westinghouse Hanford Company.

Figure 4-1

RPP-WTP DBE Horizontal Response Spectra
(0.50%, 2%, 3%, 4%, 5%, 7%, 10%, 12% Damping)

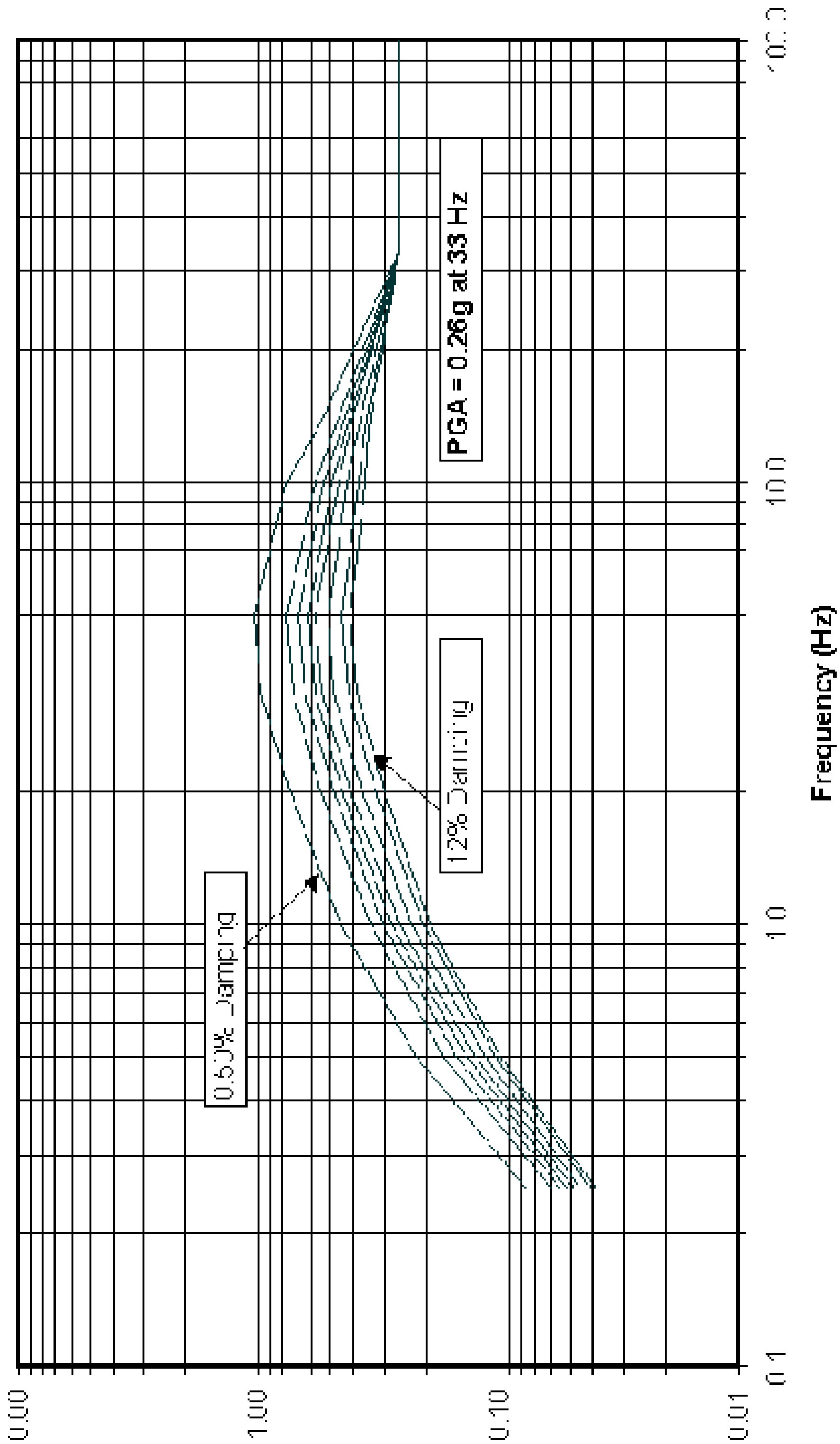
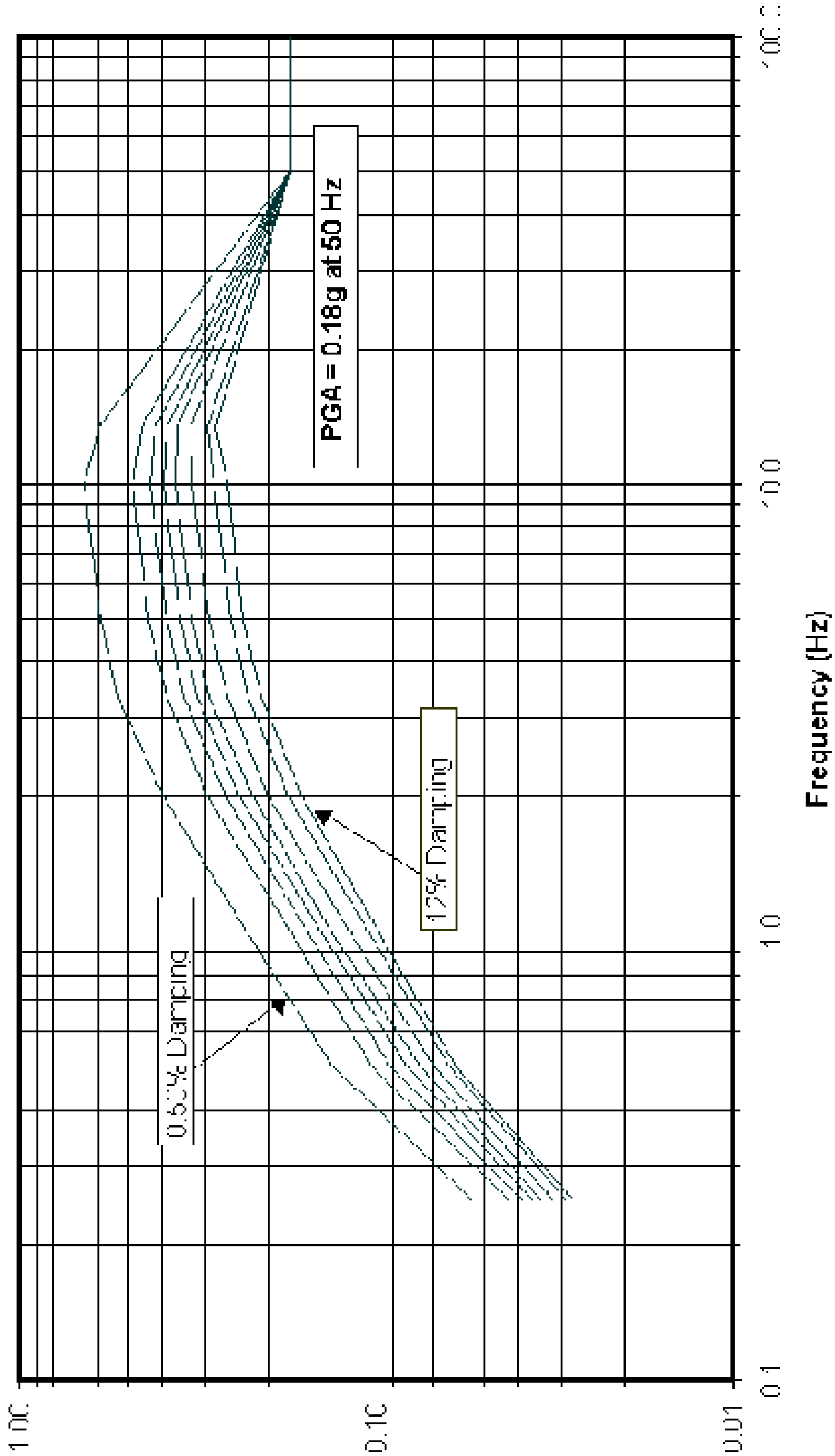


Figure 4-2

RPP-WTP DBE Vertical Response Spectra
(0.50%, 2%, 3%, 4%, 5%, 7%, 10%, 12% Damping)



Implementing Codes and Standards

ACI 349-01, *Code Requirements for Nuclear Safety-Related Concrete Structures*
ACI 349R-01, *Commentary on Code Requirements for Nuclear Safety-Related Concrete Structures*
ACI 530-99, *Building Code Requirements for Masonry Structures and Commentary*
ANSI/AISC N690-94, *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities*
ASCE 4-98, *Seismic Analysis of Safety-Related Nuclear Structures and Commentary*
ASCE 7-95, *Minimum Design Loads for Buildings and Other Structures*
DOE-STD 1020-94 (Change 1, 1996), *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*
IEEE 344-1987 (R1993), *Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations*
1997, *UBC Uniform Building Code*
DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
PCA Publication, EB 080.01, *Strength Design of Anchorage to Concrete*

Regulatory Basis

DOE/RL-96-0006 *4.2.2.2 Proven Engineering Practices/Margins-Common-Mode/Common-Cause Failure*

Safety Criterion: 4.1 - 4

This criterion addresses natural phenomena hazards (NPH) design for structures, systems, and components (SSCs) without NPH safety functions.

This criterion also addresses SSCs required to protect workers and members of the public from exposure to chemical hazards with an NPH safety function.

SSCs that may be important to the safety of the RPP-WTP shall be designed to withstand the effects of NPH such as earthquakes, wind, and floods. The SSCs included under this criterion are:

1. SSCs Important to Safety (either Safety Design Class or Safety Design Significant) that do not have an NPH safety function,
2. SSCs that are not Important to Safety and that have significant inventories of radioactive or hazardous materials but in amounts less than quantities that might lead to an Important to Safety designation, and
3. SSCs that are important to safety because of their function to protect workers and members of the public from exposure to chemical hazards.

These SSCs are designated Seismic Category III (SC-III) for earthquakes and Performance Category 2 (PC-2) for other NPH.

SSCs included under this criterion shall be designed to withstand the NPH loadings as provided in Table 4-2.

Table 4-2. Natural Phenomena Design Loads for SSCs without NPH Safety Functions

Hazard	Load	Source Document for Load
Seismic	Uniform Building Code [a], Static Force Procedure	DOE-STD-1020-94 [b]
Straight wind	91 mi/hr 3-second gust, at 33 ft above ground, Importance factor, I=1.00	DOE Newsletter [c]
Wind Missile	Not Applicable	DOE-STD-1020-94 [b]
Tornado and Tornado Missiles	Not Applicable	DOE-STD-1020-94 [b]
Volcanic ash	5 lb/ft^2	HNF-SD-GN-ER-501 [d]
Flooding	Dry site for river flooding Local Precipitation: 2.5 in. for 6 hours	HNF-SD-GN-ER-501 [d]
Snow	15.0 lb/ft^2 snow load	HNF-SD-GN-ER-501 [d]

[a] 1997, *Uniform Building Code,* International Conference of Building Officials, Whittier, California.

[b] DOE STD-1020-94, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, U.S. Department of Energy, Washington, D.C., Change 1, 1996.

[c] DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98

[d] HNF-SD-GN-ER-501, Rev. 1, "Natural Phenomena Hazards, Hanford Site, South-Central Washington", Westinghouse Hanford Company

Implementing Codes and Standards

ACI 318-99, *Building Code Requirements for Structural Concrete*
ACI 318R-99, *Commentary on Building Code Requirements for Structural Concrete*
AISC MO16-89, *Manual for Steel Construction - Allowable Stress Design, Ninth Edition*
ASCE 7-95, *Minimum Design Loads for Buildings and Other Structures*
DOE-STD 1020-94 (Change 1, 1996), *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*
1997, *UBC Uniform Building Code*
ACI 530-99, *Building Code Requirements for Masonry Structures and Commentary*
DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98
24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
PCA Publication, EB 080.01, *Strength Design of Anchorage to Concrete*

Regulatory Basis

DOE/RL-96-0006 *4.2.2.2 Proven Engineering Practices/Margins-Common-Mode/Common-Cause Failure*

Safety Criterion: 4.1 - 5

Structures, systems, and components designated as Safety Design Class shall be appropriately protected against dynamic effects (e.g., the effects of missiles, pipe whipping, and discharging fluids) that may result from failures of moderate and high energy systems or other accident conditions.

In consideration of the need to protect structures, systems, and components which are designated as Safety Design Class from these dynamic effects, the failure of the moderate or high energy system need not be postulated to occur simultaneously with an accident unless the events are causally related.

Implementing Codes and Standards

ACI 349-01, *Code Requirements for Nuclear Safety-Related Concrete Structures*
ACI 349R-01, *Commentary on Code Requirements for Nuclear Safety-Related Concrete Structures*
ANSI/AISC N690-94, *Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities*
ASCE 4-98, *Seismic Analysis of Safety-Related Nuclear Structures and Commentary*
ASCE 7-95, *Minimum Design Loads for Buildings and Other Structures*
DOE-STD 1020-94 (Change 1, 1996), *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*
DOE Newsletter (Interim Advisory on Straight Winds and Tornados) Dated 1/22/98
PCA Publication, EB 080.01, *Strength Design of Anchorage to Concrete*

Safety Criterion: 4.1 - 6

Adequate provisions for facility security and physical protection of structures, systems, and components Important to Safety shall be provided.

Implementing Codes and Standards

PL-W375-MG0004, Safeguards and Security Program Plan

Regulatory Basis

DOE/RL-96-0006 *4.3.6.1 Security-Security*

4.2 Confinement Design

Safety Criterion: 4.2 - 1

The facility shall be designed to retain the radioactive and hazardous material through a conservatively designed confinement system for normal operations, anticipated operational occurrences, and accident conditions. The confinement system shall protect the worker and public from undue risk of releases such that the radiological and chemical exposure standards of Safety Criteria 2.0-1 and/or 2.0-2 are not exceeded.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix B, Implementing Standard for Defense in Depth
24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification
DOE IG Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria, 2.3
DOE Order 420.1 Facility Safety, 4.1.1.2

Regulatory Basis

DOE/RL-96-0006 *4.1.1.4 Defense in Depth-Mitigation*

Safety Criterion: 4.2 - 2

Important to Safety liquid and gaseous systems and components, including pressure vessels, tanks, heat exchangers, piping, and valves, shall be designed to retain their hazardous inventory such that the radiological and chemical worker or public exposure standards of Safety Criteria 2.0-1 and/or 2.0-2 are not exceeded.

Implementing Codes and Standards

ASME B31.3-96 Process Piping
ASME SEC VIII Boiler and Pressure Vessel Codes, Rules for Construction of Pressure Vessels
24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Safety Criterion: 4.2 - 3

Codes and standards for Important to Safety vessels and piping should be supplemented by additional measures (such as erosion/corrosion programs and piping in-service inspections) to mitigate conditions arising that could lead to a release of radiological or chemical material that would exceed the worker or public exposure standards of Safety Criteria 2.0-1 and/or 2.0-2.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification
24590-WTP-SRD-ESH-01-001-02, Appendix E, Reliability, Availability, Maintainability, and Inspectability (RAMI)
Document P001/2 Rules for the Design of Piping Systems
Document V001/2 Rules for the Design of Vessels

Regulatory Basis

DOE/RL-96-0006 *4.2.2.4 Proven Engineering Practices/Margins-Codes and Standards*

Safety Criterion: 4.2 - 4

Liquid and gaseous storage systems designated as Important to Safety shall have continuous monitoring to detect the loss or degradation of their safe storage function. As appropriate the following shall be monitored:

1. temperature; pressure; radioactivity in ventilation exhaust and liquid effluent streams
2. liquid levels
3. tank chemistry; condensate and cooling water
4. generation of flammable and explosive mixtures of gases

Implementing Codes and Standards

ANSI N42.18-1980 (Rev 1991) Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents

IEEE 603-1991 Criteria for Safety Systems for Nuclear Power Generating Stations

ISA S12.13 PT 1-95 Performance Requirements, Combustible Gas Detectors

4.3 Engineered Safety Systems

Safety Criterion: 4.3 - 1

Engineered safety systems shall be designed (1) to initiate automatically the operation of appropriate systems to assure that specified acceptable design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of Important to Safety systems and components. The ability to manually initiate engineered safety systems shall be provided.

Implementing Codes and Standards

ANSI/ANS 58.8-1994 Time Response Design Criteria for Safety-Related Operator Actions
24590-WTP-SRD-ESH-01-001-02, Appendix B, Implementing Standard for Defense in Depth
IEEE 603-1991 Criteria for Safety Systems for Nuclear Power Generating Stations
ISA S84.01-96 Application of Safety Instrumented Systems for the Process Industries

Regulatory Basis

DOE/RL-96-0006 *4.1.1.5 Defense in Depth-Automatic Systems*

Safety Criterion: 4.3 - 2

When single failure protection is required, Important to Safety engineered safety systems shall be designed to assure that the effects of natural phenomena (including lightning), and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix B, Implementing Standard for Defense in Depth
IEEE 323-83 Qualifying Class 1E Equipment for Nuclear Power Generating Stations
IEEE 344-1987 Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
IEEE 379-1994 Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems
IEEE 384-1992 Standard Criteria for Independence of Class 1E Equipment and Circuits
NFPA 780-95 Standard for the Installation of Lightning Protection Systems
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.3 - 3

Important to Safety engineered safety systems shall be designed for high functional reliability and in-service testability commensurate with the safety functions to be performed. Design provisions should be included to limit the loss of safety functions due to damage to several structures, systems, or components Important to Safety resulting from a common-cause or common-mode failure.

The protection system shall be designed to permit periodic testing of its functioning when the facility is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

Implementing Codes and Standards

IEEE 338-1987 Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems
IEEE 379-1994 Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems

Regulatory Basis

DOE/RL-96-0006 4.2.2.2 Proven Engineering Practices/Margins-Common-Mode/Common-Cause Failure

Safety Criterion: 4.3 - 4

Important to Safety instrumentation and controls shall be provided to monitor variables and systems and control systems and components over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate public and worker safety by compliance to the standards of Safety Criteria 2.0-1 and 2.0-2, including those variables and systems that can affect the performance of Important to Safety facility conditions. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges. The instrumentation and controls provided shall provide the ability to detect off normal conditions, mitigate accidents, and place the facility in a safe state.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix B, Implementing Standard for Defense in Depth
DOE IG Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosive Safety Criteria, 2.3
DOE Order 420.1 Facility Safety, 4.1.1.2
IEEE 603-1991 Criteria for Safety Systems for Nuclear Power Generating Stations
ISA S84.01-96 Application of Safety Instrumented Systems for the Process Industries

Regulatory Basis

DOE/RL-96-0006 4.1.1.3 Defense in Depth-Control
DOE/RL-96-0006 4.2.6.2 Human Factors-Instrumentation and Control Design

Safety Criterion: 4.3 - 5

When single failure protection is required, Important to Safety protection systems shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix B, Implementing Standard for Defense in Depth
IEEE 384-1992 Standard Criteria for Independence of Class 1E Equipment and Circuits

Safety Criterion: 4.3 - 6

The possibility of human error in facility operations shall be taken into account in the design by facilitating correct decisions by operators and inhibiting wrong decisions and by providing means for detecting and correcting or compensating for error. The parameters to be monitored in control areas shall be selected and their displays arranged to ensure operators have clear and unambiguous indication of the status of the facility. The parameters and displays shall facilitate monitoring and the initiation and operation of systems designated as Important to Safety.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix B, Implementing Standard for Defense in Depth
IEEE 1023-88 Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations

Regulatory Basis

DOE/RL-96-0006	*4.1.1.6*	*Defense in Depth-Human Aspects*
DOE/RL-96-0006	*4.2.6.1*	*Human Factors-Human Error*
DOE/RL-96-0006	*4.2.6.2*	*Human Factors-Instrumentation and Control Design*
DOE/RL-96-0006	*4.2.6.3*	*Human Factors-Safety Status*

Safety Criterion: 4.3 - 7

The control room or control area shall be designed to permit occupancy and actions to be taken to monitor the facility safely during normal operations, and to provide safe control of the facility for anticipated operational occurrences and accident conditions. If credit is taken for operator action to satisfy the accident exposure standards of Safety Criteria 2.0-1 and/or 2.0-2, adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body gamma and 30 rem beta skin for the duration of the accident. For occurrences and accidents involving chemical release, provisions shall be made such that the operator exposure does not exceed the worker exposure standards of 29 CFR 1910.120 for emergency exposure.

Consideration shall also be given to accidents at nearby facilities if operator action is required to safely control the processes and bring them to a safe state.

The need for an alternate system that would allow the processes to be placed in a safe state in the event the primary control area is uninhabitable shall be evaluated.

Implementing Codes and Standards

ASME N509-89 Nuclear Power Plant Air Cleaning Units and Components
ASME N510-1989 (Rev 1995) Testing of Nuclear Air Cleaning Systems
NUREG-0800 Standard Review Plan, Section 6.4, Section II, Items 1-5.
24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis

DOE/RL-96-0006	*4.2.4.1 Emergency Preparedness-Support Facilities*
DOE/RL-96-0006	*4.2.6.2 Human Factors-Instrumentation and Control Design*
29 CFR 1910.120	*Hazardous Waste Operations and Emergency Response*

4.4 Electrical and Mechanical Systems

Safety Criterion: 4.4 - 1

A list of electric and mechanical components designated as Important to Safety shall be prepared and maintained. The list shall include:

(1) The performance specifications for normal operation and under conditions existing during and following accidents.

(2) The load, pressure, voltage, frequency, and other characteristics, as appropriate, for which the performance specified can be ensured.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

Safety Criterion: 4.4 - 2

Structures, systems, and components Important to Safety shall be designed and qualified to function as intended in the environments associated with the events for which they are intended to respond. The effects of aging on normal and abnormal functioning shall be considered in design and qualification.

Implementing Codes and Standards

10 CFR 50.49 Environmental qualification of electric equipment important to safety for nuclear power
IEEE 323-83 Qualifying Class 1E Equipment for Nuclear Power Generating Stations

Regulatory Basis

DOE/RL-96-0006 4.2.2.3 Proven Engineering Practices/Margins-Safety System Design and Qualification

Safety Criterion: 4.4 - 3

This Criterion has been deleted.

Safety Criterion: 4.4 - 4

Structures, systems, and components Important to Safety shall be designated, designed and constructed to permit appropriate inspection, testing, and maintenance throughout their operating lives to verify their continued acceptability for service with an adequate safety margin.

Systems and components designated as Important to Safety that are located in closed cells where access is not possible during facility operation or scheduled shutdown periods shall be designed and constructed to standards aimed at ensuring their suitability for the entire service life with an adequate safety margin. Alternately, provisions may be made for remote replacement, standby cells, or equipment or other methods capable of ensuring a serviceable facility with adequate safety for the duration of the intended operating life.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Appendix A, Implementing Standard for Safety Standards and Requirements Identification

24590-WTP-SRD-ESH-01-001-02, Appendix E, Reliability, Availability, Maintainability, and Inspectability (RAMI)

IEEE 338-1987 Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems

IEEE 603-1991 Criteria for Safety Systems for Nuclear Power Generating Stations

Regulatory Basis

DOE/RL-96-0006 4.2.7.1 Reliability, Availability, Maintainability, and Inspectability (RAMI)-Reliability

DOE/RL-96-0006 4.2.7.2 Reliability, Availability, Maintainability, and Inspectability (RAMI)-Availability, Maintainability, and Inspectability

Safety Criterion: 4.4 - 5

Each air treatment system designated as Safety Design Class shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and confinement capabilities to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

The use of alternate equipment may be considered to satisfy the single failure requirement.

Implementing Codes and Standards

IEEE 379-1994 Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems

IEEE 603-1991 Criteria for Safety Systems for Nuclear Power Generating Stations

Safety Criterion: 4.4 - 6

Each air treatment system designated as Safety Design Class shall be designed to ensure its operability under normal and accident conditions. The design shall permit appropriate periodic inspection and pressure and functional testing to assure:

(1) the structural and leaktight integrity of its components

(2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves

(3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems

Implementing Codes and Standards

ASME N509-89 Nuclear Power Plant Air Cleaning Units and Components
IEEE 338-1987 Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems
UL 586-90 High Efficiency Particulate Air Filter Units, Seventh Edition

Safety Criterion: 4.4 - 7

Each air treatment system designated as Safety Design Significant shall be designed to ensure its operability under normal conditions. The design shall permit appropriate periodic inspection and pressure and functional testing to assure:

(1) the structural and leaktight integrity of its components

(2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves

(3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system

Implementing Codes and Standards

UL 586-90 High Efficiency Particulate Air Filter Units, Seventh Edition

Safety Criterion: 4.4 - 8

Ventilation systems and off-gas systems must be provided where necessary to control radiological and chemical material releases and the generation of flammable and explosive gases during normal and off-normal conditions.

Implementing Codes and Standards

ARI 670-90 Fans and Blowers
ASME N509-89 Nuclear Power Plant Air Cleaning Units and Components
ASME N510-1989 (Rev 1995) Testing of Nuclear Air Cleaning Systems
ASME PTC 11-84 Performance Test Codes, Fans
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Regulatory Basis

10 CFR 835 Occupational Radiation Protection Location: 1002

Safety Criterion: 4.4 - 9

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of systems designated as Safety Design Class. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure Safety Design Class functions are maintained in the event of postulated accidents. The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their specified safety functions assuming a single failure.

Implementing Codes and Standards

IEEE 308-91 Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE 384-1992 Standard Criteria for Independence of Class 1E Equipment and Circuits
IEEE 450-1995 Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations
IEEE 484-1996 Recommended Practice for Installation Design and Installation of Large Lead Storage Batteries for Generating Stations and Substations
IEEE 485-1983 Recommended Practice for Sizing Large Lead Storage Batteries for Generating Stations and Substations
IEEE 628-1987 Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations
IEEE 741-1990 Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations
IEEE 946-1992 Design of Safety-Related DC Auxiliary Power Systems for Nuclear Power Generating Stations

Safety Criterion: 4.4 - 10

Physical and electrical separation shall be provided between diverse or redundant Safety Design Class electrical systems. Associated circuits should be avoided.

Implementing Codes and Standards

IEEE 384-1992 Standard Criteria for Independence of Class 1E Equipment and Circuits

IEEE 628-1987 Standard Criteria for the Design, Installation, and Qualification of Raceway Systems for Class 1E Circuits for Nuclear Power Generating Stations

Safety Criterion: 4.4 - 11

Electric power systems designated as Safety Design Class shall be designed to ensure their operability under normal and accident conditions. The design shall permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to periodically test:

(1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses

(2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of the offsite power system and the onsite power system

Implementing Codes and Standards

IEEE 338-1987 Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems

Safety Criterion: 4.4 - 12

Electric power systems designated as Safety Design Significant shall be designed to ensure their operability under normal conditions. The design shall permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to periodically test:

(1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses

(2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system

Implementing Codes and Standards

IEEE 338-1987 Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems
IEEE 344-1987 Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
IEEE 384-1992 Standard Criteria for Independence of Class 1E Equipment and Circuits
IEEE 387-1995 Standard Criteria for Diesel-Generator Units Applied as Standby Power Generating Stations
NFPA 70-1996 National Electric Code

Safety Criterion: 4.4 - 13

Instrument air systems designated as Safety Design Class shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming on-site power is not available) the system safety function can be accomplished, assuming a single failure.

Implementing Codes and Standards

ANS 59.3-1992 Nuclear Safety Criteria for Control Air Systems
IEEE 379-1994 Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems
IEEE 603-1991 Criteria for Safety Systems for Nuclear Power Generating Stations

Safety Criterion: 4.4 - 14

Instrument air systems designated as Safety Design Class that provide air to a non-Safety Design Class air system shall be provided with adequate isolation such that failure of the non-Safety Design Class portion of the system will not prevent the Safety Design Class portion from performing its specified safety function.

Implementing Codes and Standards

ANS 59.3-1992 Nuclear Safety Criteria for Control Air Systems

Safety Criterion: 4.4 - 15

Instrument air systems designated as Safety Design Class shall be designed to ensure their operability under normal and accident conditions. The design shall permit appropriate periodic pressure and functional testing to assure:

(1) air quality

(2) the structural integrity of its components

(3) the operability and the performance of the active components of the system

(4) the operability of the system as a whole and, under conditions as close to design as practical, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources

Implementing Codes and Standards

ANS 59.3-1992 Nuclear Safety Criteria for Control Air Systems
ASME B31.3-96 Process Piping
ASME PTC 9-70 Performance Test Codes, Displacement Compressors, Vacuum Pumps and Blowers
ASME SEC VIII Boiler and Pressure Vessel Codes, Rules for Construction of Pressure Vessels
IEEE 338-1987 Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems

Safety Criterion: 4.4 - 16

Instrument air systems designated as Safety Design Significant shall be designed to ensure their operability under normal conditions. The design shall permit appropriate periodic pressure and functional testing to assure:

(1) air quality

(2) the structural integrity of its components

(3) the operability and the performance of the active components of the system

(4) the operability of the system as a whole and, under conditions as close to design as practical, including operation of applicable portions of the protection system

Implementing Codes and Standards

ASME B31.3-96 Process Piping
ASME PTC 9-70 Performance Test Codes, Displacement Compressors, Vacuum Pumps and Blowers
ASME SEC VIII Boiler and Pressure Vessel Codes, Rules for Construction of Pressure Vessels
ISA S7.0.01-1996 Quality Standard for Instrument Air

Safety Criterion: 4.4 - 17

Instrument air systems supplying air to Important to Safety equipment shall provide clean, dry, and oil free air to this equipment. The instrument air shall be free of all corrosive and hazardous gases which may be drawn into the system.

Implementing Codes and Standards

ISA S7.0.01-1996 Quality Standard for Instrument Air

Safety Criterion: 4.4 - 18

Cooling water systems designated as Safety Design Class shall have suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to ensure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming on-site power is not available) the system safety function can be accomplished, assuming a single failure.

Implementing Codes and Standards

IEEE 379-1994 Application of the Single Failure Criterion to Nuclear Power Generating Station Safety Systems
IEEE 603-1991 Criteria for Safety Systems for Nuclear Power Generating Stations

Safety Criterion: 4.4 - 19

Cooling water systems designated as Safety Design Class shall be designed to ensure their operability under normal and accident conditions. The design shall permit appropriate periodic inspection and pressure and functional testing to assure:

(1) Long term corrosion and/or organic fouling that could degrade system performance is detected. This shall include consideration of the impacts of organic fouling on heat exchanger performance.

(2) The potential for radioactive leakage into and out of the system and to the environment is minimized.

(3) The structural and leaktight integrity of its components.

(4) The operability and the performance of the active components of the system.

(5) The operability of the system as a whole and, under conditions as close to design as practical, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Implementing Codes and Standards

ASME B31.3-96 Process Piping
ASME SEC VIII Boiler and Pressure Vessel Codes, Rules for Construction of Pressure Vessels
IEEE 338-1987 Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems

Safety Criterion: 4.4 - 20

Cooling water systems designated as Safety Design Significant shall be designed to ensure their operability under normal conditions. The design shall permit appropriate periodic inspection and pressure and functional testing to assure:

(1) Long term corrosion and/or organic fouling that could degrade system performance is detected. This shall include consideration of the impacts of organic fouling on heat exchanger performance.

(2) The potential for radioactive leakage into and out of the system and to the environment is minimized.

(3) The structural and leaktight integrity of its components.

(4) The operability and the performance of the active components of the system.

(5) The operability of the system as a whole and, under conditions as close to design as practical, including operation of applicable portions of the protection system.

Implementing Codes and Standards

ASME B31.3-96 Process Piping
ASME SEC VIII Boiler and Pressure Vessel Codes, Rules for Construction of Pressure Vessels
NFPA 214-96 Standard on Water-Cooling Towers
TEMA B, C, or R TEMA Class “B”, “C”, or “R” Heat Exchangers Mechanical Standards

Safety Criterion: 4.4 - 21

Safety Design Class motor operated valves shall be specified to ensure operability against the maximum differential pressure that might occur while performing their specified accident prevention or mitigation safety function at the minimum specified terminal voltage. Consideration for mis-positioned valves is not a requirement in determining the maximum differential pressure.

Periodic testing of Safety Design Class motor operated valves shall be performed to confirm their ability to perform their specified accident prevention or mitigation safety function.

Implementing Codes and Standards

IEEE 338-1987 Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems
IEEE 382-1985 Standard for Qualification of Actuators for Power Operated Valve Assemblies with Safety-Related Functions for Nuclear Power Plants

4.5 Fire Protection

Safety Criterion: 4.5 - 1

Two reliable and separate water supplies of adequate capacity for fire suppression shall be provided.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 2

Buildings containing a significant quantity of radioactive and/or hazardous material shall be constructed of noncombustible or fire-resistive material, where appropriate.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 3

Confinement of the fire to its origin should be achieved through passive barriers and by activating systems such as fire and smoke dampers, exhaust fans, and drainage pumps to prevent migration of gases, hot combustion products, and flammable liquids outside the fire area.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 4

Automatic fire extinguishing systems shall be included in all areas subject to loss of Safety Design Class systems, significant life safety hazards, or unacceptable program interruption, unless the Fire Hazards Analysis dictates otherwise.

As determined by the Fire Hazards Analysis special hazards shall be provided with additional fixed protection systems.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 5

Redundant Safety Design Class systems and components should be in separate fire areas.

Redundant, primary and secondary, fire protection systems shall be provided in areas where Safety Design Class systems and components are vulnerable to fire damage and where no redundant safety capability exists outside of the fire area.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 6

The design shall incorporate life safety features including means to notify and evacuate building occupants in the event of a fire, such as a fire detection or fire alarm system and illuminated, protected egress paths.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 7

The facility shall include a fire detection system to detect the presence of a fire and activate alarm systems so that measures for confinement and suppression of the fire and personnel evacuation may start promptly. The detection system shall include a means to summon the Hanford Site fire department. The system shall be capable of operation without offsite power.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 8

The facility shall include physical access and appropriate equipment to facilitate effective intervention by the Hanford Site fire department, such as an interior standpipe system.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 9

The facility design shall provide for the prevention of accidental release of significant quantities of contaminated products of combustion and fire fighting water to the environment. This can be provided by such features as ventilation control and filter systems, curbs, dikes, and holding ponds.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 10

Fire and related hazards that are unique to the facility and are not addressed by industry codes and standards shall be protected by isolation, segregation, or use of special fire control systems, such as inert gas or explosion suppression, as determined by the Fire Hazards Analysis.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 11

Fire protection systems shall be designed, and/or systems and components protected, such that its/their inadvertent operation, inactivation, or failure of structural stability will not result in the loss of a Safety Design Class function.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 12

The fire protection program shall establish the fire protection policy for the protection of the facility and the procedures, equipment, and personnel required to implement the program. The program shall have the following objectives:

(1) To prevent fires from starting

(2) To detect early, control and extinguish promptly those fires that do occur

(3) To provide protection for Safety Design Class SSCs

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 13

The fire protection program and features shall be characterized by a level of fire protection that is sufficient to fulfill the requirements of the best protected class of industrial risks (“Highly Protected Risk” or “Improved Risk”) and shall be provided protection to achieve “defense-in-depth.”

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 14

A fire protection program shall be developed that will minimize the potential for the occurrence of a fire or explosive threat and, should such an event occur, the program will limit:

(1) Radiological and hazardous releases from the facility

(2) The threat to the health and safety of facility workers

(3) Interruption of the facility mission to process tank waste

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 15

The fire protection program will include:

(1) organization, training, and responsibilities of the fire protection staff, including a trained and equipped emergency services organization

(2) inspection, testing, and maintenance of all fire protection systems by personnel properly qualified by experience and training in fire protection systems

(3) surveillance to ensure that fire barriers are in place and that fire suppression systems and components are operable

(4) training of all employees in basic fire safety

(5) periodic performance of fire drills

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 16

The fire protection program will include a plan to identify, prioritize, and monitor the status of fire protection-related appraisal findings/recommendations until final resolution is achieved. When final resolution will be significantly delayed, appropriate interim compensatory measures shall be implemented to minimize the fire risk.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 17

The fire protection program shall ensure fire protection requirements are documented and incorporated in the plans and specifications for all new facilities and for significant modifications of existing facilities. This includes a documented review by a qualified fire protection engineer of plans, specifications, procedures, and acceptance tests.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 18

The fire protection program shall include a comprehensive, documented fire protection self-assessment program, which includes all aspects (program and facility) of the fire protection program.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 19

Administrative controls shall be established to minimize fire hazards. These shall include procedures to:

(1) govern the handling and storage of combustible and flammable materials

(2) govern the handling of transient fire loads in buildings containing Safety Design Class SSCs

(3) designate staff members responsible for fire protection review of proposed work activities

(4) govern the use of ignition sources (e.g., through the use of a flame permit system)

(5) control the expedient removal of combustibles resulting from work activities

(6) establish compensatory controls for activities which may result in the impairment of fire prevention and/or mitigation features

(7) maintain periodic housekeeping inspections to ensure continued compliance with these administrative controls

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 20

A Fire Hazard Analysis (FHA) of the facility shall be performed. Such a systematic analysis shall divide the facility into "fire areas" and evaluate the fire safety of each area and of the facility as a whole. The analysis shall, for each fire area:

(1) Account for all radioactive, hazardous, and combustible materials, including estimates of their heat content

(2) Describe the processes performed and their potential for fire or explosion

(3) Account for the sources of heat and flame

(4) List the fire detection and suppression equipment

(5) Consider credible fire scenarios and evaluate the adequacy of the fire protection measures

In addition, the FHA shall consider other buildings or installations close to process buildings that contain flammable, combustible, or reactive liquid or gas storage.

The FHA shall confirm that the facility can be placed in a safe state during and after all credible fire and explosion conditions.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
DOE-STD-1066-97 Fire Protection Design Criteria
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 21

The fire protection program shall be under the direction of an individual who has been delegated authority commensurate with the responsibilities of the position and who has available staff knowledgeable in both fire protection and nuclear safety.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 22

The facility should have on file, and ready to use, a Pre-Fire Plan. The Pre-Fire Plan should assign individual and alternate responsibilities for responding to a fire alarm or call; assessing the situation, suppressing incipient fires, assembling the emergency service organization, personnel evacuation, orderly shutdown of processes, and safeguarding (if necessary) and control of radioactive and hazardous material.

The plan should clearly indicate, preferably with the help of site plans and drawings, the locations of the fire department-compatible connections and fire-fighting equipment, such as portable extinguishers, automatic fire suppression systems, sectional valves, standpipes, hydrants, and hoses. It should also indicate the areas of concentrations of combustibles, storage of flammable and combustible liquids, and areas where use of water for fire suppression is restricted because of nuclear criticality or other concerns.

The Pre-Fire Plan should be prepared in consultation and coordination with the Hanford Site fire department. The Hanford Site fire department personnel should be given familiarization tours of the facility at least once a year.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Safety Criterion: 4.5 - 23

Hot work permits shall be issued for hot work operations conducted in or near the facility. The permit shall document that applicable fire prevention and protection requirements have been implemented prior to beginning the hot work operations; it shall indicate the date(s) authorized for hot work; and identify the object on which hot work is to be performed. The permit shall be kept on file until completion of the hot work operations.

Implementing Codes and Standards

DOE G-440.1 Implementation Guide for use with DOE Orders 420.1 and 440.1 Fire Safety Program
NFPA 801-95 Standard for Facilities Handling Radioactive Materials

Regulatory Basis

DOE/RL-96-0006 5.2.8 Hot Work Control

5.0 Radiation Protection

Safety Criterion: 5.0 - 1

A Radiation Protection Program (RPP) compliant with 10 CFR 835 shall be developed and submitted for approval to DOE.

The RPP-WTP Radiological Controls Program shall address all items in 10 CFR 835 and the additional Safety Criteria provided in SRD Volume II Sections 5.1 and 5.2.

Implementing Codes and Standards

DOE G 441.1-1, Management and Administration of Radiation Protection Programs Guide

Regulatory Basis

10 CFR 835 Occupational Radiation Protection Location: 101(a-f)
DOE/RL-96-0006 4.2.3.1 Radiation Protection-Radiation Protection Practices
DOE/RL-96-0006 4.3.2.1 Radiation Protection-Radiation Practices
DOE/RL-96-0006 4.3.2.2 Radiation Protection-Procedures and Monitoring

5.1 Occupational Radiation Protection

Safety Criterion: 5.1 - 1

This safety criterion has been deleted.

Safety Criterion: 5.1 - 2

A respiratory protection program shall be established that includes:

(1) Use of respiratory protection equipment, including equipment used as emergency devices, that is tested and certified or had certification extended by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA).

(2) Air sampling sufficient to identify the potential hazard, permit proper equipment selection, and estimate exposures.

(3) Surveys and bioassays, as appropriate, to evaluate actual intakes.

(4) Testing of respirators for operability immediately prior to each use.

(5) Written procedures regarding selection, fitting, issuance, maintenance, and testing of respirators, including testing for operability immediately prior to each use; supervision and training of personnel; monitoring, including air sampling and bioassays; and recordkeeping.

(6) Determination by a physician prior to the initial fitting of respirators, and either every 12 months thereafter or periodically at a frequency determined by a physician, that the individual user is medically fit to use the respiratory protection equipment.

(7) A written policy statement on respirator usage covering:

   (i) The use of process or other engineering controls, instead of respirators.

   (ii) The routine, nonroutine, and emergency use of respirators.

   (iii) The periods of respirator use and relief from respirator use. Each respirator user will be informed that they may leave the area at any time for relief from respirator use in the event of equipment malfunction, physical or psychological distress, procedural or communication failure, significant deterioration of operating conditions, or any other conditions that might require such relief.

(8) Use of equipment within limitations for type and mode of use and provision for proper visual, communication, and other special capabilities (such as adequate skin protection) when needed.

(9) Notification to the Regulator, in writing, at least 30 days before the date that respiratory protection equipment is first used to protect workers from airborne radioactivity.

Implementing Codes and Standards

ANSI Z-88.2-1992 American National Standard for Respiratory Protection

Safety Criterion: 5.1 - 3
This safety criterion has been deleted.

Safety Criterion: 5.1 - 4
This safety criterion has been deleted.

Safety Criterion: 5.1 - 5
This safety criterion has been deleted.

Safety Criterion: 5.1 - 6
This safety criterion has been deleted.

Safety Criterion: 5.1 - 7
This safety criterion has been deleted.

5.2 Occupational Radiation Protection Design

Safety Criterion: 5.2 - 1

This Safety Criterion has been deleted

Safety Criterion: 5.2 - 2

This Safety Criterion has been deleted

Safety Criterion: 5.2 - 3

This Safety Criterion has been deleted

Safety Criterion: 5.2 - 4

This Safety Criterion has been deleted

5.3 Environmental Radiation Protection

Safety Criterion: 5.3 - 1

An Environmental Radiological Protection Program shall be prepared and submitted to the regulator.

The Environmental Radiological Protection Program (ERPP) shall address the following elements, as appropriate:

(1) the identity of existing and anticipated types of activities and areas of the site subject to the ERPP

(2) the measures to be used to implement the ERPP

(3) the methods to be used to monitor, report, and record compliance with the ERPP

(4) models and methods used for dose assessment including bioaccumulation and dose-conversion factors

(5) an As Low As is Reasonably Achievable (ALARA) Program

(6) effluent and environmental monitoring including:

- (i) sources of airborne emissions
- (ii) sources of discharges in liquid waste streams
- (iii) effluent monitoring
- (iv) environmental surveillance
- (v) meteorological data acquisition
- (vi) pre-operational evaluation

(7) ground water protection

(8) radiological protection in the management of radioactive waste

(9) controls on the release of materials

(10) property containing residual radioactive materials

Implementing Codes and Standards

ANSI/ISO-14001-1996, Environmental Management Systems - Specifications with guidance for use

Regulatory Basis

DE-AC06-96RL13308	*Part I Section C.5 Table S4-1*	
DOE/RL-96-0006	*4.3.2.1*	*Radiation Protection-Radiation Practices*
DOE/RL-96-0006	*4.3.2.2*	*Radiation Protection-Procedures and Monitoring*

Safety Criterion: 5.3 - 2

The ALARA Program shall ensure that releases of radioactive materials to the environment and exposures to the public during normal operations shall be kept ALARA and within prescribed limits.

Implementing Codes and Standards

DOE G 441.1-2, Occupational ALARA Program Guide

Regulatory Basis

DOE/RL-96-0006 3.2 Radiation Protection Objective
DOE/RL-96-0006 4.2.3.2 Radiation Protection-Radiation Protection Features
WAC 173-480 Ambient Air Quality Standards and Emission Limits for Radionuclides Location: Part 050 (1)

Safety Criterion: 5.3 - 3

A waste management program shall ensure compliance with all applicable laws and regulations. The waste management program shall also ensure that the radiological impact to the general public and environment due to radioactive wastes arising from RPP-WTP operation shall be ALARA.

Implementing Codes and Standards

IAEA Safety Series No. 50-SG-011, Operational Management for Radioactive Effluents and Wastes Arising in Nuclear Power Plants.
ANSI/ISO-14001-1996, Environmental Management Systems - Specifications with guidance for use

Regulatory Basis

DOE/RL-96-0006 3.2 Radiation Protection Objective
DOE/RL-96-0006 4.2.3.2 Radiation Protection-Radiation Protection Features

Safety Criterion: 5.3 - 4

Equipment shall be designed and installed to monitor and maintain control over radioactive materials in gaseous and liquid effluents produced during normal operations, including anticipated operational occurrences.

Implementing Codes and Standards

40 CFR 52 Appendix E Performance Specifications and Specification Test Procedures for Monitoring Systems for Effluent Stream Gas Volumetric Flow Rate
40 CFR 60 Appendix A, Methods 1, 1a, 2, 2a, 2c, 2d, 4, 5,and 17
ANSI N13.1-1969 (R 1993) Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities
ANSI N42.18-1980 (R 1991) Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents
ANSI N323 Radiation Protection Instrumentation Test and Calibration
ASME/ANSI AG-1, Code on Nuclear Air and Gas Treatment
ASME/ANSI N509, Nuclear Power Plant Air-Cleaning Units and Components
ASME/ANSI N510, Testing of Nuclear Air Cleaning Systems
ACGIH 1988, Industrial Ventilation, A Manual of Recommended Practice, 20th Edition, American Conference of Governmental Industrial Hygienists.
ERDA 76-21, Nuclear Air Cleaning Handbook

Regulatory Basis

DOE/RL-96-0006 3.2 Radiation Protection Objective
DOE/RL-96-0006 4.2.3.2 Radiation Protection-Radiation Protection Features
WAC 246-247 Radiation Protection - Air Emissions Location: Part 075
WAC 246-247 Radiation Protection - Air Emissions Location: Part 110

Safety Criterion: 5.3 - 5

All new construction and significant modifications of air emission units shall utilize best available radionuclide control technology (BARCT).

Implementing Codes and Standards

WAC 246-247-120 Appendix B BARCT Compliance Demonstration
ASME/ANSI AG-1, Code on Nuclear Air and Gas Treatment
ASME/ANSI N509, Nuclear Power Plant Air-Cleaning Units and Components
ASME/ANSI N510, Testing of Nuclear Air Cleaning Systems
ANSI N13.1, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities
ANSI N42.18, Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents
ERDA 76-21, Nuclear Air Cleaning Handbook
ACGIH 1988, Industrial Ventilation, A Manual of Recommended Practice, 20th Edition, American Conference of Governmental Industrial Hygienists.
40 CFR 60 Appendix A, Methods 1, 1a, 2, 2a, 2c, 2d, 4, 5,and 17

Regulatory Basis

WAC 173-480 Ambient Air Quality Standards and Emission Limits for Radionuclides Location: Part 060
WAC 246-247 Radiation Protection - Air Emissions Location: Part 040 (3)

Safety Criterion: 5.3 - 6

Activities shall be conducted in such a manner that no radioactive material is discharged into sanitary sewers. Exempt from this Safety Criterion are trace radioactive materials present in:

(1) readily soluble waste such as kitchen waste from breakrooms, custodial cleaning solutions, or other materials of similar non-RPP-WTP process origin

(2) biological waste (solid and liquid human waste) which is readily dispersed in water

Also exempt from this Safety Criterion are excreta from individuals undergoing medical diagnosis or therapy with radioactive materials.

Implementing Codes and Standards

IAEA Safety Series No. 50-SG-011, Operational Management for Radioactive Effluents and Wastes Arising in Nuclear Power Plants.

Regulatory Basis

DOE/RL-96-0006	*3.2*	*Radiation Protection Objective*
DOE/RL-96-0006	*4.2.3.2*	*Radiation Protection-Radiation Protection Features*

Safety Criterion: 5.3 - 7

Liquid discharges from the facility, other than sanitary sewer discharges, shall comply with ALARA process requirements, be treated by the best available technology, and not result in release of settleable solids to surface waters for streams exceeding 5 pCi/g for alpha-emitting radionuclides, and/or 50 pCi/g for beta-emitting radionuclides.

Note: The RPP-WTP design does not include provisions for liquid waste discharges, other than sanitary sewer discharges. Therefore, Implementing Codes and Standards are not required. If the RPP-WTP design changes such that liquid discharges result, an SRD revision will be prepared.

Safety Criterion: 5.3 - 8

Controls on the release of materials and property containing residual radioactive material shall be established.

Implementing Codes and Standards

10 CFR 835, Occupational Radiation Protection, Appendix D (ad hoc)

Note: The Appendix D values will be used as surface contamination criteria for determining the suitability of releasing material from radiologically controlled areas. These criteria are not applicable to materials potentially contaminated throughout their volume. Because the RPP-WTP process feed is a mixed waste, any items that are determined to be contaminated, will also be assumed to be a mixed waste (i.e., containing a State of Washington dangerous waste). Rather than determine the quantities of dangerous wastes present, these materials will be disposed of as mixed wastes.

Regulatory Basis

DOE/RL-96-0006	*3.2*	*Radiation Protection Objective*
DOE/RL-96-0006	*4.2.3.1*	*Radiation Protection-Radiation Protection Practices*

5.4 Environmental Radiological Monitoring

Safety Criterion: 5.4 - 1

Each source shall have capability for independent effluent emission testing as follows:

(1) Sampling ports adequate for test methods applicable to each source

(2) Safe sampling platform(s)

(3) Safe access to sampling platform(s)

(4) Utilities for sampling and testing equipment

(5) Any other facilities deemed necessary to safely and properly test a source

Implementing Codes and Standards

ANSI N13.1-1969 (R 1993) Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities

Regulatory Basis

40 CFR 61 National Emission Standards for Hazardous Air Pollutants Location: 13
WAC 246-247 Radiation Protection - Air Emissions Location: Part 075 (10)
WAC 246-247 Radiation Protection - Air Emissions Location: Part 075 (9)

Safety Criterion: 5.4 - 2

Nonpoint and fugitive emissions of radioactive material shall be monitored.

Implementing Codes and Standards

ANSI N13.1-1969 (R 1993) Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities

Regulatory Basis

WAC 246-247 Radiation Protection - Air Emissions Location: Part 075 (8)

Safety Criterion: 5.4 - 3

Direct measurements shall be made, to the extent practicable, to obtain information characterizing source terms, exposures, exposure modes, and other information needed in evaluating doses.

Implementing Codes and Standards

ANSI N13.1-1969 (R 1993) Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities

Regulatory Basis

WAC 246-221 Radiation Protection Standards Location: 070 (1)

Safety Criterion: 5.4 - 4

When the effluents from a single source, or from two or more sources subject to the same emission standards, are combined before being released to the atmosphere, a monitoring system shall be installed on each effluent or on the combined effluent. If two or more sources are not subject to the same emission standards, a separate monitoring system shall be installed on each effluent. If the applicable standard is a mass emission standard and the effluent from one source is released to the atmosphere through more than one point, a monitoring system shall be installed at each emission point.

Implementing Codes and Standards

ANSI N13.1-1969 (R 1993) Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities

Regulatory Basis

40 CFR 61 National Emission Standards for Hazardous Air Pollutants Location: 14 (d)

Safety Criterion: 5.4 - 5

Equipment and procedures used for the continuous monitoring of radioactive air emissions shall conform, to applicable guidance.

Implementing Codes and Standards

40 CFR 52 Appendix E Performance Specifications and Specification Test Procedures for Monitoring Systems for Effluent Stream Gas Volumetric Flow Rate
40 CFR 60 Appendix A, Test Methods 1, 1a, 2, 2a, 2c, 2d, 4, 5,and 17
40 CFR 61 Appendix B, Test Method 114
ANSI N13.1-1969 (R 1993) Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities
ANSI N323 Radiation Protection Instrumentation Test and Calibration
ANSI N42.18-1980 (R 1991) Specification and Performance of On-Site Instrumentation for Continuously Monitoring Radioactivity in Effluents

Regulatory Basis

40 CFR 61 National Emission Standards for Hazardous Air Pollutants Location: 93
WAC 246-247 Radiation Protection - Air Emissions Location: Part 075 (2)

Safety Criterion: 5.4 - 6

Computer codes or procedures used to determine the offsite total effective dose equivalent from airborne emissions shall be EPA approved.

Implementing Codes and Standards

ANSI/ISO-14001-1996, Environmental Management Systems - Specification with Guidance for Use

Regulatory Basis

40 CFR 61 National Emission Standards for Hazardous Air Pollutants Location: 93
WAC 246-247 Radiation Protection - Air Emissions Location: Part 085 (2)

Safety Criterion: 5.4 - 7

Compliance with the annual dose limit for individual members of the public (100 mrem/yr from all sources) shall be shown by:

(1) Demonstrating by measurement or calculation that the total effective dose equivalent to the individual likely to receive the highest dose from the operation does not exceed the annual dose limit; or

(2) Demonstrating that:

(a) The annual average concentrations of radioactive material released in gaseous and liquid effluents at the boundary of the unrestricted area do not exceed the values specified in Table II of WAC246-221-290.

(b) If an individual were continuously present in an unrestricted area, the dose from external sources would not exceed 0.002 rem in an hour and 0.05 rem in a year.

Implementing Codes and Standards

ANSI/ISO-14001-1996, Environmental Management Systems - Specification with Guidance for Use

Regulatory Basis

40 CFR 61 National Emission Standards for Hazardous Air Pollutants Location: 93
WAC 246-221 Radiation Protection Standards Location: 070 (2)
WAC 246-247 Radiation Protection - Air Emissions Location: Part 085 (1)

Safety Criterion: 5.4 - 8

Compliance with the public air emission standard shall be determined by calculating the highest effective dose equivalent to any member of the public at any offsite point where there is a residence, school, business or office.

The determination of compliance shall include all radioactive air emissions resulting from routine and nonroutine operations for the past calendar year.

Implementing Codes and Standards

ANSI/ISO-14001-1996, Environmental Management Systems - Specification with Guidance for Use

Regulatory Basis

40 CFR 61 National Emission Standards for Hazardous Air Pollutants Location: 94
WAC 246-247 Radiation Protection - Air Emissions Location: Part 085 (3)

Safety Criterion: 5.4 - 9

Records sufficient to demonstrate compliance with the dose limit for individual members of the public shall be maintained. Records must document the source of input parameters including the results of all measurements upon which they are based, the calculations and/or analytical methods used to derive values for input parameters, and the procedure used to determine compliance. This documentation should be sufficient to allow an independent auditor to verify the accuracy of the determination made concerning the facility's compliance.

Implementing Codes and Standards

ANSI/ISO-14001-1996, Environmental Management Systems - Specification with Guidance for Use

Regulatory Basis

40 CFR 61 National Emission Standards for Hazardous Air Pollutants Location: 95
WAC 246-247 Radiation Protection - Air Emissions Location: Part 080

Safety Criterion: 5.4 - 10

An environmental surveillance program shall be developed and implemented to include:

(1) Meteorological data acquisition (Note 1)

(2) Pre-operational evaluation (Note 2)

(3) Near-Facility Monitoring (Note 3)

(4) Ground Water Protection (Note 4)

Implementing Codes and Standards

ANSI/ISO-14001-1996, Environmental Management Systems - Specification with guidance for use

IAEA Safety Series No 41, *Objectives and Design of Environmental Monitoring Programmes for Radioactive Contaminants*

Notes:

1. BNFL-5193-ID-03, *Interface Control Document*, Revision 2, *ICD-22 between DOE and BNFL Inc. for Air Emissions,* Table 2 states that DOE will maintain the Hanford Site Air Operating Permit (AOP) and provide access to meteorological data.
2. BNFL-5193-ID-03, *Interface Control Document*, Revision 2, *ICD-09 Between DOE and BNFL Inc. for Land Siting,* Table 1, describes specific interfaces responsibilities for the RPP-WTP contractor and for the DOE. Item 12 of the table requires that the RPP-WTP contractor perform any additional site characterization work beyond that which was performed by the DOE. The RPP describes the plans and measures for compliance with the survey and contamination control requirements of 10 CFR 835.
3. As described in BNFL-5193-ID-03, *Interface Control Document*, Revision 2, *ICD-22 between DOE and BNFL Inc. for Air Emissions,* DOE will continue to operate site and near-facility monitoring networks in the vicinity of the RPP-WTP site. Additional monitoring which is required will be provided by the RPP-WTP contractor. If additional monitoring is required, it will be performed consistent with the Hanford Site near-facility monitoring program for inclusion in site annual reports (example, HNF-EP-0573-6, *Hanford site Near-Facility Environmental Monitoring Annual Report, Calendar Year 1997*).
4. BNFL-5193-ID-03, *Interface Control Document*, Revision 2, *ICD-09 between DOE and BNFL Inc. for Land Siting,* Section 3.3, Ground Water Monitoring Wells, states that that the DOE will "…close groundwater monitoring well E25-32 prior to the start of site work…" There is no liquid discharge to the environment from RPP-WTP operations. Transfer piping to the Effluent Treatment Facility is by means of a three-inch pipe encased in a 6-inch pipe. Potential leakage from the transfer pipe is contained, and collected by the outer pipe. Accidental release of the inner pipe contents would be detected by the transfer pipe leak detection equipment. If both inner and outer pipes failed, such leakage could result in soil contamination which would be remediated prior to any contamination reaching the ground water.

Regulatory Basis

DOE/RL-96-0006	*3.2*	*Radiation Protection Objective*
DOE/RL-96-0006	*4.2.3.1*	*Radiation Protection-Radiation Protection Practices*

6.0 Startup

Safety Criterion: 6.0 - 1

A pre-operational testing program shall be established and followed to demonstrate that Important to Safety structures, systems and components have been properly constructed and can perform their specified functions. The program shall provide for the detection, tracking, and correction of deficiencies.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.14 Startup Testing
Section: 3.14 Startup Testing and Operation
Section: 5.6.4 Startup Review

Regulatory Basis

DOE/RL-96-0006 *4.2.8.1 Pre-Operational Testing-Testing Program*

Safety Criterion: 6.0 - 2

Procedures for normal facility and systems operation and for functional tests to be performed during the operating phase shall be validated as part of the pre-operational testing program.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.14 Startup Testing
Section: 3.14 Startup Testing and Operation
Section: 5.6.4 Startup Review

Regulatory Basis

DOE/RL-96-0006 *4.2.8.2 Pre-Operational Testing-Operational Systems and Functional Testing Procedures Validation*

Safety Criterion: 6.0 - 3

During pre-operational testing, detailed diagnostic data shall be collected on systems and components designated as Important to Safety and the initial operating parameters of the systems and components shall be recorded.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.14 Startup Testing
Section: 3.14 Startup Testing and Operation
Section: 5.6.4 Startup Review

Regulatory Basis

DOE/RL-96-0006 *4.2.8.3 Pre-Operational Testing-Safety Systems Data*

Safety Criterion: 6.0 - 4

During the pre-operational testing program, the as-built operating characteristics of process systems, and systems and components designated as Important to Safety shall be determined and documented. Operating points shall be adjusted to conform to values in the design basis. Training procedures and limiting conditions for operation shall be modified, if necessary, to accurately reflect the operating characteristics of the systems and components as built.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.14 Startup Testing
Section: 3.14 Startup Testing and Operation
Section: 5.6.4 Startup Review

Regulatory Basis

DOE/RL-96-0006 *4.2.8.4* *Pre-Operational Testing-Design Operating Characteristics*

Safety Criterion: 6.0 - 5

A pre-startup safety review shall be performed. The pre-startup safety review shall confirm that, prior to the introduction of radioactive or process chemicals considered to pose a hazard to a process, construction and equipment is in accordance with design specifications; safety, operating, maintenance, and emergency procedures are in place and are adequate; a process hazard analysis has been performed and recommendations have been resolved or implemented before startup; and training of each employee involved in operating a process has been completed.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.14 Startup Testing
Section: 5.6.4 Startup Review

Regulatory Basis

DOE/RL-96-0006 *4.3.1.4* *Conduct of Operations-Readiness*
DOE/RL-96-0006 *5.2.6* *Pre-startup Safety Review*

7.0 Management and Operations

Safety Criterion: 7.0 - 1

Normal operations shall be conducted in accordance with approved operational safety requirements and in strict accordance with administrative and procedural controls.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.13 Procedures
Section: 5.6.1 Procedure Development

Regulatory Basis

DOE/RL-96-0006 4.3.1.2 Conduct of Operations-Normal Operations
DOE/RL-96-0006 5.1.3 Process Safety Responsibility

Safety Criterion: 7.0 - 2

Normal operation, including anticipated operational occurrences, maintenance, and testing, shall be controlled so that facility and system variables remain within their normal operating ranges and the frequency of demands placed on Important to Safety structures, systems, and components are small.

Implementing Codes and Standards

Regulatory Basis

DOE/RL-96-0006 4.1.1.3 Defense in Depth-Control

Safety Criterion: 7.0 - 3

The operating organizations shall become and remain familiar with the features and limitations of components included in the design of the facility. They shall obtain appropriate input from the design organization on pre-operational testing, operating procedures, and the planning and conduct of training.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.14 Startup Testing
Section: 1.3.15 Operations

Regulatory Basis

DOE/RL-96-0006 4.1.5.2 Configuration Management-Contractor Design Knowledge

Safety Criterion: 7.0 - 4

The assignment and subdivision of responsibility for safety within the contractor's organization shall be kept well defined throughout the life of the facility.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.2 Safety Responsibilities
Section: 6.1.2 Lines of Authority and Responsibility

Regulatory Basis

DOE/RL-96-0006 *4.1.2.2 Safety Responsibility-Safety Assignments*
DOE/RL-96-0006 *4.3.1.1 Conduct of Operations-Organizational Structure*

7.1 Management and Organization/Staffing

Safety Criterion: 7.1 - 1

Safety management shall oversee the implementation of the risk management program elements. A qualified person shall be assigned the overall responsibility for the development, implementation, and integration of the risk management program elements. If responsibility for implementing individual requirements of the risk management program is assigned to other persons, the names or positions of these people shall be documented and the lines of authority defined through an organization chart or similar document.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 6.1 Integration into Work Planning and Performance

Regulatory Basis

Safety Criterion: 7.1 - 2

When selecting a subcontractor, information regarding the subcontract employer's safety performance and programs shall be obtained and evaluated. Subcontract employees shall be informed of the known potential fire, explosion, or toxic release hazards related to the subcontractor's work and the process. The applicable provisions of the emergency plan shall be explained to the subcontractors. Safe work practices to control the entrance, presence, and exit of subcontract employers and employees in radioactive or hazardous process areas shall be developed and implemented. The performance of subcontract employers with regard to safety shall be periodically evaluated and a subcontract employee injury and illness log related to the subcontractor's work in process areas shall be maintained.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 5.2 Control of Subcontractors

Regulatory Basis

DOE/RL-96-0006 5.2.5 Subcontractors

Safety Criterion: 7.1 - 3

A framework shall be established for safety review organizations that are responsible for assuring the safety of the facility. The separation between the responsibilities of the safety review organizations and those of the other organizations shall remain clear so that the safety review organizations retain their independence as safety authorities. Internal safety oversight should be conducted by qualified personnel to ensure that the safety standards are consistently met.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.13 Procedures
Section: 3.16 Internal Safety Oversight
Chapter: 10.0 Assessments

Regulatory Basis

DOE/RL-96-0006	*4.1.4.1*	*Safety/Quality Culture-Safety/Quality Culture*
DOE/RL-96-0006	*4.3.1.5*	*Conduct of Operations-Internal Surveillance and Audits*
DOE/RL-96-0006	*4.4.1*	*Safety Review Organization*
DOE/RL-96-0006	*4.4.2*	*Qualified Personnel*
DOE/RL-96-0006	*5.1.3*	*Process Safety Responsibility*

Safety Criterion: 7.1 - 4

Commitments from outside organizations to provide data and services required to satisfy safety obligations shall be made prior to the need for the information or services.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.18 Emergency Planning
Section: 3.10 Emergency Preparedness

Regulatory Basis

DOE/RL-96-0006	*4.1.2.3*	*Safety Responsibility-Site and Technical Support*

7.2 Training and Procedures

Safety Criterion: 7.2 - 1

Programs providing for continual training and qualification for operations, maintenance, and technical support personnel, to enable them to perform their duties safely and efficiently, shall be developed and implemented utilizing a tailored approach.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.12 Training
Section: 3.15 Training and Qualification
Section: 4.2.2 Training and Procedures

Regulatory Basis

DOE/RL-96-0006 *4.3.1.7* *Conduct of Operations-Access to Technical Safety Support*
DOE/RL-96-0006 *4.3.4.2* *Training and Qualifications-Training Programs*

Safety Criterion: 7.2 - 2

A training program must be established, implemented, and maintained for individuals relied upon to operate, maintain, or modify the facility in a safe manner. The training program shall be based on a systems approach to training that includes the following:

(a) Systematic analysis of the jobs to be performed

(b) Learning objectives derived from the analysis which describe desired performance after training

(c) Training design and implementation based on the learning objectives

(d) Evaluation of trainee mastery of the objectives during training

(e) Evaluation and revision of the training based on the performance of trained personnel in the job setting

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.15 Training and Qualification

Safety Criterion: 7.2 - 3

Each employee involved in operating a process shall be trained in an overview of the process and in the operating procedures/instructions. The training shall include emphasis on the specific safety and health hazards, operating limits, emergency operations including shutdown, and safe work practices applicable to the employee's job tasks.

Refresher training shall be provided at least every three years, and more often if necessary, to each employee involved in operating a process to assure that the employee understands and adheres to the current operating procedures/instructions of the process and is proficient in the procedures to follow if conditions exceed the design basis of the facility.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 5.6.3 Development of the Operator Training Program

Regulatory Basis

DOE/RL-96-0006 4.3.4.1 Training and Qualifications-Personnel Training
DOE/RL-96-0006 4.3.4.3 Training and Qualifications-Conditions Beyond Design Basis
DOE/RL-96-0006 5.2.4 Training
WAC 246-247 Radiation Protection - Air Emissions Location: Part 075 (12)

Safety Criterion: 7.2 - 4

Up-to-date records of training status shall be maintained which contain the names of the trained employees, the types of training, the dates of training, and the means used to verify that the employees understood the training.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.15 Training and Qualification
Chapter: 8.0 Document Control and Maintenance

Regulatory Basis

Safety Criterion: 7.2 - 5

Written procedures/instructions that provide clear direction for safely conducting activities involving radioactive or hazardous materials shall be developed and implemented for each phase of the facility life. The procedures/instructions shall address at least the following elements:

(1) Steps for each operating phase:
- (a) Initial startup
- (b) Normal operations
- (c) Temporary operations
- (d) Emergency shutdown including the conditions under which emergency shutdown is required, and the assignment of shutdown responsibility to qualified operators to ensure that emergency shutdown is executed in a safe and timely manner
- (e) Emergency operations
- (f) Normal shutdown
- (g) Startup following a turnaround, or after an emergency shutdown

(2) Operating limits:
- (a) Consequences of deviation
- (b) Steps required to correct or avoid deviation

(3) Safety and health considerations:
- (a) Properties of, and hazards presented by, the chemicals used in the process
- (b) Precautions necessary to prevent exposure, including engineering controls, administrative controls, and personal protective equipment
- (c) Control measures to be taken if physical contact or airborne exposure occurs
- (d) Quality control for raw materials and control of hazardous chemical inventory levels
- (e) Any special or unique hazards

(4) Safety systems and their functions

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 5.6.1 Procedure Development

Regulatory Basis

DOE/RL-96-0006	*4.3.2.2*	*Radiation Protection-Procedures and Monitoring*
DOE/RL-96-0006	*5.2.3*	*Operating Procedures*

Safety Criterion: 7.2 - 6

Operating procedures shall be readily accessible to employees who work in or maintain a process with radioactive or hazardous materials.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 5.6.1 Procedure Development

Regulatory Basis

Safety Criterion: 7.2 - 7

Operating procedures shall be reviewed as often as necessary to assure that they reflect current operating practice, including changes that result from changes in process chemicals, technology, and equipment, and changes to facilities. These procedures shall be certified annually that they are current and accurate.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 5.3 Configuration Management
Section: 5.6.1 Procedure Development

Regulatory Basis

Safety Criterion: 7.2 - 8

Safe work practices providing for the control of hazards during operations such as lockout/tagout; confined space entry; opening process equipment or piping; and control over entrance into a facility by maintenance, subcontractor, laboratory, or other support personnel shall be developed. These safe work practices shall apply to employees and subcontractor employees.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.4 Safety/Quality Culture
Section: 5.2 Control of Subcontractors
Section: 5.6.6 Hot Work Operations

Regulatory Basis

7.3 Quality Assurance Program

Safety Criterion: 7.3 - 1

The following quality assurance elements shall be applied using a graded approach:

(1) Quality Assurance Program
(2) Personnel Training and Qualification
(3) Quality Improvement
(4) Documents and Records
(5) Work Processes
(6) Design
(7) Procurement
(8) Inspection and Acceptance Testing
(9) Management Assessment
(10) Independent Assessment.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.11 Quality Levels

Regulatory Basis

10 CFR 830.120	*Quality assurance requirements*	*Location: (b)(1)*
DOE/RL-96-0006	*4.1.6.1 Quality Assurance-Quality Assurance Application*	
DOE/RL-96-0006	*4.1.6.2 Quality Assurance-Established Techniques and Procedures*	

Safety Criterion: 7.3 - 2

A written Quality Assurance Program (QAP) shall be developed, implemented, and maintained. The QAP shall describe the organizational structure, functional responsibilities, levels of authority, and interfaces for those managing, performing, and assessing the work. The QAP shall describe management processes, including planning, scheduling, and resource considerations.

Implementing Codes and Standards

ASME NQA-1-1989
Section 2, “Quality Assurance Program” (including associated supplements)

Regulatory Basis

10 CFR 830.120	*Quality assurance requirements*	*Location: (a)(1)*
10 CFR 830.120	*Quality assurance requirements*	*Location: (c)(1)(i)*
DOE/RL-96-0006	*4.1.1.6 Defense in Depth-Human Aspects*	
DOE/RL-96-0006	*4.1.4.1 Safety/Quality Culture-Safety/Quality Culture*	
WAC 246-247	*Radiation Protection - Air Emissions*	*Location: Part 075 (6)*

Safety Criterion: 7.3 - 3

Personnel shall be trained and qualified to ensure they are capable of performing their assigned work. Personnel shall be provided continuing training to ensure that job proficiency is maintained.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.15 Training and Qualification

Regulatory Basis

10 CFR 830.120 *Quality assurance requirements* *Location: (c)(1)(ii)*

Safety Criterion: 7.3 - 4

Documents shall be prepared, reviewed, approved, issued, used, and revised to prescribe processes, specify requirements, or establish design. Records shall be specified, prepared, reviewed, approved, and maintained.

Implementing Codes and Standards

ASME NQA-1-1989
Section 3S-1, 7, “Documentation and Records” (including associated supplements)
Section 3S-1, 17, “Quality Assurance Records” (including associated supplements)

Regulatory Basis

10 CFR 830.120 *Quality assurance requirements* *Location: (c)(1)(iv)*

Safety Criterion: 7.3 - 5

Work shall be performed to established technical standards and administrative controls using approved instructions, procedures, or other appropriate means. Items shall be identified and controlled to ensure their proper use. Items shall be maintained to prevent their damage, loss, or deterioration. Equipment used for process monitoring or data collection shall be calibrated and maintained.

Implementing Codes and Standards

ASME NQA-1-1989
Section 3, “Design Control” (including associated supplements)
Section 5, “Instructions, Procedures, and Drawings” (including associated supplements)
Section 8, “Identification and Control of Items” (including associated supplements)
Section 9, “Control of Processes” (including associated supplements)
Section 12, “Control of Measuring and Test Equipment” (including associated supplements)

Regulatory Basis

10 CFR 830.120 *Quality assurance requirements* *Location: (c)(2)(i)*
DOE/RL-96-0006 *4.1.6.3 Quality Assurance-Established Techniques and Procedures*

Safety Criterion: 7.3 - 6

Processes to detect and prevent quality problems shall be established and implemented. Items, services, and processes that do not meet established requirements shall be identified, controlled, and corrected. Correction shall include identifying the causes of problems and preventing recurrence. Item characteristics, process implementation, and other quality-related information shall be reviewed and the data analyzed to identify items, services, and processes needing improvement.

Implementing Codes and Standards

ASME NQA-1-1989
Section 15, "Control of Nonconforming Items" (including associated supplements)
Section 16, "Corrective Action" (including associated supplements)

Regulatory Basis

10 CFR 830.120 *Quality assurance requirements* *Location: (c)(1)(iii)*
DOE/RL-96-0006 *4.1.4.1 Safety/Quality Culture-Safety/Quality Culture*

Safety Criterion: 7.3 - 7

Inspection and testing of specified items, services, and processes shall be conducted using established acceptance and performance criteria. Equipment used for inspections and tests shall be calibrated and maintained.

Implementing Codes and Standards

ASME NQA-1-1989
Section 10, "Inspections" (including associated supplements)
Section 11, "Test Control" (including associated supplements)
Section 12, "Control of Measuring and Test Equipment" (including associated supplements)

Regulatory Basis

10 CFR 830.120 *Quality assurance requirements* *Location: (c)(2)(iv)*

Safety Criterion: 7.3 - 8

Managers shall assess their management processes. Problems that hinder the organization from achieving its objectives shall be identified and corrected.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Chapter: 10.0 Assessments

Regulatory Basis

10 CFR 830.120 *Quality assurance requirements* *Location: (c)(3)(i)*

Safety Criterion: 7.3 - 9

Independent assessment shall be planned and conducted to measure item and service quality, to measure the adequacy of work performance, and to promote improvement. The group performing independent assessments shall have sufficient authority and freedom from the line to carry out its responsibilities. Persons conducting independent assessments shall be technically qualified and knowledgeable in the areas assessed.

Implementing Codes and Standards

ASME NQA-1-1989
Section 18, "Audits" (including associated supplements)

Regulatory Basis

10 CFR 830.120 *Quality assurance requirements* *Location: (c)(3)(ii)*

Safety Criterion: 7.3 - 10

Compliance audits shall be performed at least every three years to verify that the procedures and practices developed to ensure nuclear and process safety are adequate and are being followed. The compliance audit shall be conducted by at least one person knowledgeable in the process. A report of the findings of the audit shall be developed. An appropriate response shall be determined and documented for each of the findings of the compliance audit, and it shall be documented when deficiencies have been corrected. Employers shall retain the two most recent compliance audit reports.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 5.4 Compliance Audits
Chapter: 8.0 Document Control and Maintenance

Regulatory Basis

DOE/RL-96-0006 *5.2.12* *Compliance Audits*

Safety Criterion: 7.3 - 11

Procured items and services shall meet established requirements and perform as specified. Prospective suppliers shall be evaluated and selected on the basis of specified criteria. Processes to ensure that approved suppliers continue to provide acceptable items and services shall be established and implemented.

Implementing Codes and Standards

ASME NQA-1-1989
Section 7, "Control of Purchased Items and Services" (including associated supplements)

Regulatory Basis

10 CFR 830.120 *Quality assurance requirements* *Location: (c)(2)(iii)*

Safety Criterion: 7.3 - 12

Changes made to the Quality Assurance Program (QAP) shall be submitted annually to the regulator for review. The submittal shall identify the changes, the pages affected, the reason for the changes, and the basis for concluding that the revised QAP continues to satisfy the requirements of this section.

Regulatory Basis

10 CFR 830.120 *Quality assurance requirements* *Location: (b)(3)*

7.4 Unreviewed Safety Questions

Safety Criterion: 7.4 - 1

A safety evaluation shall be performed to determine whether a situation involves an unreviewed safety question (USQ) for:

(1) Temporary or permanent changes in the facility as described in the existing authorization basis

(2) Temporary or permanent changes in the procedures as derived from existing authorization basis

(3) Tests or experiments not described in the existing authorization basis

A situation involves a USQ if:

1) the probability of occurrence or the radiological or chemical consequences of an accident or malfunction of equipment Important to Safety, previously evaluated in the facility safety analyses or other related safety analysis and evaluations not yet included in the updated facility analysis, may be increased
2) a possibility for an accident or equipment malfunction of a different type than any evaluated previously in the facility safety analyses or other related safety analysis and evaluations not yet included in the updated facility safety analysis, may be created
3) any margin of safety is reduced

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.16.4 Unreviewed Safety Questions

Regulatory Basis

DOE/RL-96-0006	*4.4.4*	*Unresolved Safety Questions*
DOE/RL-96-0006	*5.2.9*	*Management of Change*

Safety Criterion: 7.4 - 2

Regulatory approval shall be obtained for situations determined to involve an unreviewed safety question or a change in a technical safety requirement, prior to initiating the activity, if the initiation of the activity would itself involve a USQ, or implementing the proposed change.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.16.4 Unreviewed Safety Questions

Regulatory Basis

DOE/RL-96-0006	*4.4.4*	*Unresolved Safety Questions*

Safety Criterion: 7.4 - 3

Procedures shall be developed and implemented to govern the need for, and the performance of, safety evaluations.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.16.4 Unreviewed Safety Questions

Safety Criterion: 7.4 - 4

When information is identified that indicates a potential inadequacy of previous analyses as defined in the approved Safety Analysis Report (SAR) or that indicates a possible reduction in safety margins as defined in the technical safety requirements, the following actions shall occur:

(1) Notify the regulator of the situation upon discovery of the information

(2) Perform a safety evaluation

(3) Take action as appropriate to place or maintain the facility in a safe condition until the safety evaluation is completed

(4) Submit the completed safety evaluation prior to removing any operational restrictions initiated

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.16.3 Incident Investigations

Safety Criterion: 7.4 - 5

For all safety evaluations required under this section, the following actions shall occur:

(1) Document the basis for the USQ(s) determination.

(2) Incorporate in the existing SAR and technical safety requirements for the facility any changes that are needed as a result of the safety evaluation or any action taken pursuant to this section.

(3) Submit to DOE, on a schedule corresponding to the updates of the SAR for the facility, a report summarizing all situations for which a safety evaluation was required by this section and indicating all “changes” considered in a safety evaluation and implemented 3 months or more before the submittal date of the safety analysis report.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.3.3 Changes to Safety Documentation
Section: 3.16.4 Unreviewed Safety Questions

7.5 Conduct of Operations

Safety Criterion: 7.5 - 1

A program for conduct of operations at the facility shall be established and implemented using a tailored approach.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.15 Operations

Safety Criterion: 7.5 - 2

The conduct of operations program shall address:

(1) Operations organization and administration
(2) Shift routines and operating practices
(3) Control area activities
(4) Communications
(5) Control of on-shift training
(6) Investigation of abnormal events
(7) Notifications
(8) Control of equipment and system status
(9) Lockout and tagout
(10) Independent verification
(11) Logkeeping
(12) Operations turnover
(13) Operations aspects of facility chemistry and unique processes
(14) Required reading
(15) Timely orders to operators
(16) Operations procedures
(17) Operator aid postings
(18) Equipment and piping labeling
(19) Emergency operating procedures for dealing with responses to accident conditions.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.15 Operations

Regulatory Basis

DOE/RL-96-0006 *4.3.1.1* *Conduct of Operations-Organizational Structure*
DOE/RL-96-0006 *4.3.1.3* *Conduct of Operations-Emergency Operating Procedures*
DOE/RL-96-0006 *4.3.1.4* *Conduct of Operations-Readiness*

7.6 Maintenance

Safety Criterion: 7.6 - 1

A maintenance program for the facility shall be developed and implemented using a tailored approach.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 4.2.1 Engineered Features

Safety Criterion: 7.6 - 2

The maintenance program shall contain provisions sufficient to preserve, predict, and restore the availability, operability, and reliability of structures, systems, and components designated as Important to Safety.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.13 Reliability, Availability, Maintainability, and Inspectability (RAMI)

Regulatory Basis

DOE/RL-96-0006 *4.3.5.1 Operational Testing, Inspection, and Maintenance*

Safety Criterion: 7.6 - 3

The maintenance program for Important to Safety Structures, systems and components shall clearly define:

(1) The Important to Safety structures, systems, and components that comprise the facility

(2) The requirements of the maintenance program that are derived from the program elements listed in Safety Criterion 7.6-4

(3) The management systems used for those activities, including the means for monitoring and measuring the effectiveness of the program and the management of maintenance backlog

(4) The assignment of responsibilities and authority for all levels of the maintenance organization

(5) Mechanisms to feedback such relevant information as trend analysis and instrumentation performance/reliability data in order to identify necessary program modifications

(6) Provisions for identifying and evaluating possible component, system design, occupational safety and health, or other relevant problems and implementation of a self-assessment program

(7) Performance indicators and criteria to be utilized to measure equipment, systems, and personnel effectiveness in maintenance activities

(8) Interfaces between maintenance and other organizations (e.g., involving operations, engineering, quality, and safety)

(9) Quantitative reliability target values for systems and components to start or run, when such values are credited in safety analysis

(10) Appropriate authorization is received before modification starts on a safety instrumented system

(11) Assessment of impact of the modification on the functionality of the safety instrumented system is performed, to ensure functionality is not impaired

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
- Section: 1.3.10 Classification of Structures, Systems, and Components
- Section: 3.2 Safety Responsibilities
- Section: 3.4 Safety/Quality Culture
- Section: 3.13 Reliability, Availability, Maintainability, and Inspectability (RAMI)
- Section: 3.16.3 Incident Investigations
- Section: 3.16.5 Performance Monitoring
- Section: 3.16.6 Performance Indicators
- Section: 3.16.8 Feedback and Trending
- Chapter: 10.0 Assessments

24590-WTP-SRD-ESH-01-001-02, Attachment A, Implementing Standard for Safety Standards and Requirements Identification

Regulatory Basis

DOE/RL-96-0006 4.2.7.1 Reliability, Availability, Maintainability, and Inspectability (RAMI)-Reliability
DOE/RL-96-0006 4.3.5.1 Operational Testing, Inspection, and Maintenance-Operational Testing, Inspection, and Maintenance

Safety Criterion: 7.6 - 4

The maintenance program shall address each of the following elements:

(1) Organization and administration

(2) Maintenance training and qualification

(3) Maintenance facilities, equipment, and tools

(4) Types of maintenance

(5) Maintenance procedures and other work-related documents

(6) Planning, scheduling, and coordinating maintenance activities

(7) Control of maintenance activities

(8) Post-maintenance testing

(9) Procurement of parts, materials, and services

(10) Material receipt, inspection, handling, storage, retrieving, and issuance

(11) Control and calibration of measuring and test equipment

(12) Maintenance tools and equipment control

(13) Documented facility condition inspections to identify and address aging effects

(14) Management involvement with facility operations

(15) Maintenance history and trending

(16) Analysis of maintenance-related problems

(17) Modification work.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan

Section: 1.3.11 Quality Levels
Section: 4.2.2 Training and Procedures
Section: 5.3 Configuration Management
Section: 5.4 Compliance Audits
Section: 5.6.5 Mechanical Integrity
Chapter: 11.0 Organization Roles, Responsibilities, and Authorities

Regulatory Basis

DOE/RL-96-0006 4.3.5.1 Operational Testing, Inspection, and Maintenance
DOE/RL-96-0006 5.2.7 Mechanical Integrity
WAC 246-247 Radiation Protection - Air Emissions Location: Part 075 (12)

7.7 Reporting and Incident Investigation

Safety Criterion: 7.7 - 1

Each incident which resulted in, or could reasonably have resulted in a major accident shall be investigated. An incident investigation shall be initiated as promptly as possible, but not later than 48 hours following the incident.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.17 Incident Investigations
Section: 5.6.7 Investigations of Incidents

Regulatory Basis

DOE/RL-96-0006 5.2.10 Incident Investigation

Safety Criterion: 7.7 - 2

An incident investigation team shall be established and consist of at least one person knowledgeable in the process involved, including a subcontract employee if the incident involved work of the subcontractor, and other persons with appropriate knowledge and experience to thoroughly investigate and analyze the incident. A report shall be prepared at the conclusion of the investigation which includes at a minimum:

(1) Date of incident

(2) Date investigation began

(3) A description of the incident

(4) The factors that contributed to the incident

(5) Any recommendations resulting from the investigation

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.17 Incident Investigations
Section: 5.6.7 Investigations of Incidents
Chapter: 8.0 Document Control and Maintenance

Regulatory Basis

DOE/RL-96-0006 5.2.10 Incident Investigation

Safety Criterion: 7.7 - 3

A system shall be established to promptly address and resolve the incident report findings and recommendations. Resolutions and corrective actions shall be documented. The report shall be submitted to the regulator for evaluation and in support of regulatory oversight. The report shall be reviewed with all affected personnel whose job tasks are relevant to the incident findings including subcontract employees where applicable. Incident investigation reports shall be retained for five years.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 5.6.7 Investigations of Incidents

Regulatory Basis

DOE/RL-96-0006 *5.2.10* *Incident Investigation*

Safety Criterion: 7.7 - 4

The Facility Manager shall categorize reportable incidents as soon as reasonably possible and in all cases within 2 hours of identification of the event or condition. If categorization is not clear, the occurrence shall be conservatively categorized at the higher level being considered. The occurrence categorization shall be elevated, maintained, or lowered, as appropriate, as further information is obtained.

Reportable occurrences shall be categorized in accordance with the following guidance:

"Emergencies" are the most serious reportable occurrences and they require an increased alert status for on-site personnel and, in specified cases, for off-site authorities. Emergencies require a time-urgent notification as part of the facility's comprehensive emergency management program.

"Unusual Occurrences" are the category of non-emergency reportable occurrences that exceed the off-normal occurrence threshold and have significant impact or potential for impact on safety, the environment, health, safeguards and security, or operations.

"Off-Normal Occurrences" are the category of abnormal or unplanned reportable occurrences that adversely affect, potentially affect, or are indicative of degradation in the level of safety, safeguards and security, environmental or health protection, performance or operation of the facility.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 5.6.7 Investigations of Incidents

Safety Criterion: 7.7 - 5

The Facility Manager shall orally notify the regulator of Emergencies within 15 minutes. In addition, the Facility Manager shall orally notify the regulator of Unusual Occurrences, as soon as practical, but in all cases within 2 hours of categorization. The Facility Manager shall contact the regulator via the Headquarters Emergency Operations Center.

Follow-up Notification. The Facility Manager shall provide follow-up oral notification to the regulator via the Headquarters Emergency Operations Center within two hours of identification of further degradation in the level of safety of the facility or other worsening conditions. A change in a reporting level to an Emergency level is to be orally reported within 15 minutes of the recategorization.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.17 Incident Investigations

Safety Criterion: 7.7 - 6

The Facility Manager shall provide written notification of all reportable occurrences, including summaries of Emergencies, to the regulator before the close of the next business day from the time of categorization of the occurrence (not to exceed 80 hours). An Update Occurrence Report shall be submitted when significant new information (including any changes in categorization) is available.

Completion of the Final Occurrence Report is required when the analysis of the occurrence has been completed, root cause and contributing cause(s) finalized, corrective action(s) determined and scheduled, and lessons learned identified. The Final Occurrence Report shall be submitted to the regulator within 45 days of categorization of the occurrence.

The Occurrence Report shall contain, at a minimum, the following information about each reportable occurrence at the facility:

(1) An alphanumeric occurrence report number identifying the regulatory field office, the contractor, the facility, the year of the occurrence, and the sequential number of the occurrence

(2) The number of occurrences in the report

(3) Category of the occurrence

(4) Regulatory program office responsible for the facility involved

(5) All systems, equipment, structural items, administrative controls, or procedures involved in the occurrence or defect identification, including information related to manufacturers, types, model numbers, size, procedure numbers, and (for defect identification) the number of affected components

(6) Date and time when the occurrence was discovered and categorized

(7) Date and time of notification of regulatory and state and local authorities

(8) A thorough and complete description of what occurred or the defect reported, including a sequence of events and a description of failures (or potential failures and the safety significance with respect to defect identification), so that regulatory and contractor personnel, not familiar with the facility-specific design features or administrative controls employed, can understand the complete occurrence and any substantial safety hazard which could result

(9) Operational status of the facility or equipment at the time of the occurrence, including the status of structures, systems, or components that were inoperable at the start of the event and that contributed to the event

(10) Immediate or remedial actions taken to place the facility, system, or equipment in a safe, stable condition, to return them to service, or to correct or alleviate the anomalous condition, and the results of those actions

(11) Cause of the occurrence, including direct and contributing causes and the root cause

(12) Recommendations about whether further evaluation is required and, if so, before or after returning the facility to operation

(13) Action taken or planned to correct the problem and the identified cause and to prevent recurrence

(14) Impact of the occurrence on the environment, health and safety of workers, the public, and on-site and off-site environs (including quantities and types of radioactive materials released)

(15) Levels and types of contamination, human exposures, and known or projected environmental, safety, and health impacts

(16) Impact of the occurrence on the affected program and/or project

(17) Impact of the occurrence on the adequacy of national codes and standards and regulatory requirements

(18) Lessons learned from the occurrence that could be of importance to other facility operators or that shall be addressed in personnel training or facility procedures

(19) Any previous similar events at the same facility that are known to the Facility Manager

(20) The name and telephone number of a person within the Facility Manager's organization who is knowledgeable about the occurrence

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.17 Incident Investigations

Regulatory Basis

DOE/RL-96-0006 *4.3.1.8 Conduct of Operations-Operational Events*

Safety Criterion: 7.7 - 7

Procedures shall be written and implemented to carry out categorization, notification, and reporting requirements for reportable occurrences.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.16.3 Incident Investigations
Section: 5.6.7 Investigations of Incidents

Safety Criterion: 7.7 - 8

The services of the Environment Safety & Health (ES&H) reporting system, maintained by the Office of Environment, Safety, and Health, shall be used to meet the documentation and distribution requirements of this section.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.16.3 Incident Investigations

Safety Criterion: 7.7 - 9

The RPP-WTP contractor shall ensure that subcontractors and suppliers report defective items, materials, and services and shall specify these requirements in applicable procurement documents.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 5.2 Control of Subcontractors

7.8 Emergency Preparedness

Safety Criterion: 7.8 - 1

An emergency response program shall be developed, documented, and implemented for the purpose of protecting public health and the environment. The program shall include the following elements:

(1) An emergency response plan.

(2) Emergency Planning Implementing Procedures to ensure the timely and effective implementation of the provisions of the emergency plan.

(3) A facility emergency response organization, with clearly defined roles, responsibilities and authorities.

(4) A training program that provides initial and annual refresher training for facility response personnel, general employees, and response personnel from other agencies.

(5) Program administration to include maintenance of technical support documents, plans, and procedures, the coordination of activities, and maintenance of appropriate auditable records.

(6) Adequate emergency facilities and equipment to support response.

(7) The scope of the program will be designed to be commensurate with the hazards present at the facility and will be determined by performing an assessment of the hazards.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.10 Emergency Preparedness
Chapter: 8.0 Document Control and Maintenance

Regulatory Basis

WAC 246-247 Radiation Protection - Air Emissions *Location: Part 075 (12)*

Safety Criterion: 7.8 - 2

The Emergency Management Program will be documented in an emergency plan which describes the provisions for responses to Operational Emergencies. The emergency response plan will address the following program elements

(1) The establishment and maintenance of a facility emergency response organization with clearly specified authorities and responsibilities for emergency response and mitigation.

(2) Provisions for interfaces and coordination with Hanford Site and offsite agencies in the areas of planning, preparedness, response, and recovery.

(3) A description of the hazards and potential consequences resulting from analyzed accidents.

(4) Identify and describe the capabilities for the detection of emergency events, the methodology for determining event severity and the basis for declaring an emergency.

(5) The methods to be used to provide notification of an emergency event to Hanford Site organizations, offsite response agencies, and Federal, state and local regulatory agencies.

(6) Provisions for assessing the consequences resulting from the release of hazardous materials.

(7) A description of protective actions for responders, workers, and the public, to include provisions for sheltering, evacuation, and personnel accountability.

(8) Medical support during emergency response, to include provisions for ambulance/hospital services and decontamination of injured personnel.

(9) Methodology for the safe-shut down of the facility, reentry to the facility during or after emergency response and provisions for developing a recovery strategy following an accident.

(10) A public information program designed to provide the public, media and employees with accurate and timely information.

(11) A training program will de designed to ensure that personnel are prepared to respond to, manage, mitigate, and recover from emergencies associated with facility operations.

(12) Provisions for the administration of the program, to include a designated program administrator, program assessment and issue resolution, the development and maintenance of technical support documents, plans, and procedures, the coordination of activities, and maintenance of appropriate auditable records.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.10 Emergency Preparedness

Regulatory Basis

DOE/RL-96-0006	*4.1.2.3*	*Safety Responsibility-Site and Technical Support*
DOE/RL-96-0006	*5.2.11*	*Emergency Planning and Response*

Safety Criterion: 7.8 - 3

Emergency plans shall be prepared before the startup of the facility, and shall be exercised periodically to ensure that protection measures can be implemented in the event of an accident that results in, or has the potential for, unacceptable releases of radioactive materials within and beyond the facility control perimeter.

A determination shall be made of the size of the geographic area surrounding the facility, known as the Emergency Planning Zone (EPZ), within which special planning and preparedness activities will be performed to reduce the potential health and safety impacts from an event involving hazardous materials. The extent of planning and preparedness necessary shall correspond to the type and scope of hazards present and the potential consequences of events.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.18 Emergency Planning
Section: 3.10 Emergency Preparedness

Regulatory Basis

DOE/RL-96-0006 *4.3.3.3* *Emergency Preparedness-Establishment and Continued Exercise of Emergency Plans*

Safety Criterion: 7.8 - 4

The results of analyses of the facility response to accidents with the potential for releases resulting in doses in excess of Environmental Protection Agency and the State of Washington emergency clean-up standards, beyond the RPP-WTP controlled area boundary shall be used in preparing emergency operating procedures which will contain specific instructions for facility operations personnel on the shutdown of facility processes and the mitigation of accidents for all identified off-normal and emergency conditions.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.10 Emergency Preparedness

Regulatory Basis

DOE/RL-96-0006 *4.3.3.2* *Emergency Preparedness-Accident Management Strategy*

Safety Criterion: 7.8 - 5

The emergency response plan shall be coordinated with the DOE Hanford Site and local community emergency response plans.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.10 Emergency Preparedness

Regulatory Basis

DOE/RL-96-0006 *4.3.3.1* *Emergency Preparedness-Offsite Measures*

8.0 Deactivation and Decommissioning

Safety Criterion: 8.0 - 1

There shall be an approved plan for deactivation of the facility before it is constructed. The objectives of the plan shall be to reduce radiation exposure to Hanford Site personnel and the public both during and following deactivation and decommissioning activities and to minimize the quantity of radioactive waste generated during deactivation, decontamination, and decommissioning. Features and procedures that simplify and facilitate decommissioning shall be identified during the planning and design phase based upon a proposed decommissioning method.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document Volume II
Appendix F, Ad Hoc Implementing Standard for Deactivation and Decommissioning Planning

Regulatory Basis

DOE/RL-96-0006 4.2.3.3 Radiation Protection-Deactivation, Decontamination, and Decommissioning Design
DOE/RL-96-0006 4.3.2.3 Radiation Protection-Final Deactivation Plans and Provisions

Safety Criterion: 8.0 - 2

Facilities shall be designed to simplify decontamination and decommissioning, reduce exposure to site personnel and the public during these activities, and increase the potential for reuse. Features and procedures that simplify and facilitate decontamination, decommissioning, and minimization of contaminated equipment and the generation of radioactive waste during deactivation, decontamination, and decommissioning shall be identified during the planning and design phase based upon a proposed decommissioning method or conversion to other use.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, Safety Requirements Document Volume II
Appendix F, Ad Hoc Implementing Standard for Deactivation and Decommissioning Planning
DOE G 441.1-2, Occupational ALARA Program Guide

Regulatory Basis

10 CFR 835 Occupational Radiation Protection Location: 1002
DOE/RL-96-0006 4.2.3.3 Radiation Protection-Deactivation, Decontamination, and Decommissioning Design
DOE/RL-96-0006 4.3.2.3 Radiation Protection-Final Deactivation Plans and Provisions

9.0 Documentation and Submittals

Safety Criterion: 9.0 - 1

An annual report shall be prepared and submitted to the Regulator and appropriate local officials. The annual report shall document:

(1) the quantity of each of the principal radionuclides in excess of background released to unrestricted areas in liquid and gaseous effluents during the previous year of operation.

(2) the calculated annual dose to the maximally exposed members of the public, and (3) the calculated collective dose to members of the public from exposures to RPP-WTP radiation sources.

Reports on dose estimates to the public shall include relevant site-specific information, including the locations of members of the public subject to the greatest potential exposures, the population distribution subject to exposures from RPP-WTP activities, and exposure pathways germane to the site. Values of assumed default or site-specific parameters used in calculations shall be discussed and included with the documentation of the calculations.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 9.2 Scheduling of Events for Regulatory Submittals

Regulatory Basis

40 CFR 61	*National Emission Standards for Hazardous Air Pollutants*	*Location: 104*
40 CFR 61	*National Emission Standards for Hazardous Air Pollutants*	*Location: 94*

Safety Criterion: 9.0 - 2

The Contractor should request authorization for construction only after being satisfied by appropriate internal assessments that the main safety issues have been satisfactorily resolved and that the remainder are amenable to solution before operations are scheduled to begin.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 9.2 Scheduling of Events for Regulatory Submittals

Regulatory Basis

DOE/RL-96-0006	*4.4.3*	*Recommendation for Initiation of Construction*

Safety Criterion: 9.0 - 3

The results of the pre-startup safety review should be submitted to DOE for evaluation and in support of authorization decisions and regulatory oversight.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan

Section: 9.2 Scheduling of Events for Regulatory Submittals

Regulatory Basis

DOE/RL-96-0006 *5.2.6* *Pre-startup Safety Review*

Safety Criterion: 9.0 - 4

Material that is part of the authorization basis shall be established, documented, and submitted to the DOE for evaluation and in support of decisions and regulatory oversight. The material shall be maintained current with respect to changes made to the facility design and administrative controls and in the light of significantly new safety information.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan

Section: 3.3.1 Content of the Authorization Basis

Section: 3.3.2 Control of the Authorization Basis

Section: 3.3.3 Changes to Safety Documentation

9.1 Safety Analysis Reports

Safety Criterion: 9.1 - 1

Safety analyses shall be performed using a tailored approach to develop and evaluate the adequacy of the authorization basis for the facility. Preliminary and Final Safety Analysis Reports shall be prepared to document the safety analyses.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix A, "Implementing Standard for Safety Standards and Requirements Identification"
Appendix G, "Ad Hoc Implementing Standard for Safety Analysis Reports"[CJD1]

Regulatory Basis

DOE/RL-96-0006 *4.1.3.1 Authorization Basis-Authorization Basis*
DOE/RL-96-0006 *4.2.1.3 Design-Safety Analysis*

Safety Criterion: 9.1 - 2

A SAR shall contain sections that address the following topics:

(1) Site Description
(2) Facility and Process Description
(3) Integrated Safety Analysis
(4) Nuclear Criticality Safety
(5) Technical Safety Requirements
(6) Radiation Safety
(7) Chemical Safety
(8) Fire Safety
(9) Human Factors
(10) Emergency Preparedness
(11) Management Organization
(12) Conduct of Operations
(13) Procedures
(14) Training and Qualification
(15) Deactivation and Decommissioning
(16) Incident Investigations
(17) Records Management
(18) Audits and Assessments
(19) Quality Assurance
(20) Initial Surveillance and In-Service Testing
(21) Maintenance

The SAR should also contain an Executive Summary that provides an overview of the facility safety basis and presents information sufficient to establish a top-level understanding of the facility, its' operation, and the results of the safety analysis.[CJD2]

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*

Appendix G, "Ad Hoc Implementing Standard for Safety Analysis Reports"[CJD3]

Safety Criterion: 9.1 - 3

A Preliminary Safety Analysis Report (PSAR) shall be submitted to the regulator only after all major safety issues have been resolved and other safety issues scheduled for completion. The PSAR shall document the facility design and plans for construction and demonstrate adequate planning for the operational phase.

A Final Safety Analysis Report (FSAR) shall be submitted to the regulator for approval prior to authorization to operate the facility. The FSAR shall document the completed design and construction and provide details on the plans for operation. The FSAR shall include facility and process drawings and fabrication and construction specifications important to the safety analysis of the facility. The FSAR shall identify significant changes made in the facility design and plans for operation from what was presented in the PSAR.

Implementing Codes and Standards

24590-WTP-SRD-ESH-01-001-02, *Safety Requirements Document Volume II*
Appendix G, "Ad Hoc Implementing Standard for Safety Analysis Reports"[CJD4]

Safety Criterion: 9.1 - 4

The FSAR shall be reviewed annually and updated as necessary to ensure that the information is current, remains applicable, and reflects all changes implemented up to 3 months prior to the filing of the updated FSAR. The regulatory approval of any Unreviewed Safety Questions, and the material submitted to the regulator in support of that approval, shall be considered an addendum to the FSAR until the information is incorporated into the FSAR as part of the next periodic update.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.3.3 Changes to Safety Documentation

Regulatory Basis

DOE/RL-96-0006 4.1.3.1 Authorization Basis-Authorization Basis

Safety Criterion: 9.1 - 5

The SAR shall be maintained as a controlled document.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Chapter: 8.0 Document Control and Maintenance

Regulatory Basis

DOE/RL-96-0006 4.1.3.1 Authorization Basis-Authorization Basis

Safety Criterion: 9.1 - 6

All responsibilities concerning the facility as identified in the approved SAR shall be carried out.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.2 Safety Responsibilities

Safety Criterion: 9.1 - 7

The hazard analysis shall be submitted for approval as part of the SAR.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Chapter: 9.0 Scheduling of Safety-Related Activities

Regulatory Basis

9.2 Technical Safety Requirements

Safety Criterion: 9.2 - 1

Technical safety requirements shall be prepared and submitted for approval, and the facility shall be operated in accordance with the approved technical safety requirements.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.15 Operations
Section: 3.3.1.4 Technical Safety Requirements (TSR)

Regulatory Basis

DOE/RL-96-0006 4.1.3.1 Authorization Basis-Authorization Basis

Safety Criterion: 9.2 - 2

Technical safety requirements shall be based on the Final Safety Analysis Report and any facility-specific commitments made.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.3.1.4 Technical Safety Requirements (TSR)
Section: 4.2.3.4 Technical Safety Requirements and Licensee Controlled Requirements

Safety Criterion: 9.2 - 3

Technical safety requirements shall consist of the following:

(1) Safety Limits (SLs). This section of the technical safety requirements shall contain the safety limits on the process variables. If an SL for a nuclear facility is exceeded, the nuclear facility shall be immediately placed in the most stable, safe condition attainable, including total shutdown, except where such action might reduce the margin of safety. The safe, stable condition entered as a corrective action shall be maintained until DOE authorizes further operations.

(2) Operating Limits:

(a) Limiting Control Settings (LCSs). This section of the technical safety requirements shall contain the settings for automatic alarms and for the automatic or non-automatic initiation of protective actions related to those variables associated with the function of Safety Design Class structures, systems, or components (SSCs) (if the safety analyses show that they are relied upon to mitigate or prevent an accident). The specific settings chosen must correct a situation automatically or manually such that the related SL is not exceeded. If an automatic alarm or protective device does not function as required during an applicable operating mode, the contractor shall take action as defined in the LCS to maintain the variables within the requirements and to repair the automatic device promptly or to shut down the affected part of the process and, if required, to shut down the facility.

(b) Limiting Conditions for Operation (LCOs). This section of the technical safety requirements shall contain the limits on functional capability or performance level. If a limiting condition for operation is not met, the contractor shall take any remedial actions specified by the technical safety requirements until the condition can be met.

The LCOs will be based upon:

(a) Process variables, design features, and operating restrictions that are the initial conditions for accident analysis.

(b) Structures, systems, and components that must function to prevent or mitigate accidents to achieve compliance to offsite radiological and chemical exposure standards of Safety Criteria 2.0-1 and 2.0-2.

(3) Surveillance Requirements (SRs). This section of the technical safety requirements shall contain the surveillance requirements necessary to maintain operation of the facility within the SLs, LCSs, and LCOs. If a required surveillance is not successfully completed or is not performed with its required frequency, the systems or components involved shall be assumed to be inoperable and actions defined by LCSs or LCOs shall be taken until the systems or components can be shown to be operable.

(4) Administrative Controls. This section of the technical safety requirements shall contain the requirements associated with administrative controls, including those for reporting violations of the technical safety requirements. Staffing requirements for facility positions important to safe operation of the facility shall be included in the administrative controls section of the technical safety requirements. Commitments to the safety management programs identified in the SAR as necessary components of the facility safety basis shall be provided in this section.

(5) Use and Application. This section of the technical safety requirements shall contain the basic instructions for applying the safety restrictions contained in the technical safety requirements. Definitions of terms, operating modes, frequency notations, and actions to be taken in the event of violations of OLs or SRs are to be included in this section.

(6) Appendices.

(a) Basis. This appendix shall provide summary statements of the reasons for the OLs and associated SRs. The basis shall show how the numeric value, the condition, or the surveillance fulfills the purpose derived from the safety documentation. The primary purpose for describing the basis of each requirement is to ensure that any future changes to the requirement will not affect its original intent or purpose.

(b) Design Features. This appendix shall describe design features of the facility that, if altered or modified, would have a significant effect on safe operation. If design features are described in a regulator-approved SAR pursuant to this part, this appendix is not required.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.3.1.4 Technical Safety Requirements (TSR)

Regulatory Basis

DOE/RL-96-0006 *4.3.1.6 Conduct of Operations-Operations Within the Authorization Basis*

Safety Criterion: 9.2 - 4

Technical safety requirements shall be kept current at all times so that they reflect the facility as it exists and as it is analyzed in the SAR.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.3.1.4 Technical Safety Requirements (TSR)
Section: 4.2.3.4 Technical Safety Requirements and Licensee Controlled Requirements

Regulatory Basis

DOE/RL-96-0006 *4.1.3.1 Authorization Basis-Authorization Basis*

Safety Criterion: 9.2 - 5

All proposed revisions to technical safety requirements, excluding its bases, shall be submitted for regulatory approval prior to implementation of the revision. The submission shall include the basis for the proposed revision.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 3.3.3 Changes to Safety Documentation

Safety Criterion: 9.2 - 6

Only current approved technical safety requirements shall be used for the operation of the facility. Technical safety requirements shall be maintained as a controlled document with all users designated as authorized users.

Implementing Codes and Standards

24590-WTP-ISMP-ESH-01-001, Integrated Safety Management Plan
Section: 1.3.15 Operations
Section: 3.3.1.4 Technical Safety Requirements (TSR)
Section: 4.2.3.4 Technical Safety Requirements and Licensee Controlled Requirements
Chapter: 8.0 Document Control and Maintenance

9.3 Risk Management Plan (this section has been deleted)

Safety Criterion: 9.3 - 1

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Safety Criterion: 9.3 - 2

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Safety Criterion: 9.3 - 4

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Safety Criterion: 9.3 - 5

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Appendix A

Implementing Standard for Safety Standards and Requirements Identification

CONTENTS

{ TOC \o “1-3” \b “appA” }

1.0 Introduction

This standard implements the process for establishing a set of radiological, nuclear, and process safety requirements and standards as described in DOE/RL-96-0004 and RL/REG-98-17. The Project refers to this process as Integrated Safety Management (ISM).

The activities described below establish radiological, nuclear and process safety standards and requirements for design, construction, and operation of the facility. Establishment of safety standards and requirements is an iterative process that takes place throughout the life of the project. As the design evolves, the process repeatedly evaluates these standards and requirements based on the evolving design.

The Safety Requirements Document (SRD) provides formal documentation of the standards, which are a result of this process. The SRD is updated as required to reflect the results of successive iterations of the standards and requirements identification process (i.e., the ISM process).

2.0 Process Initiation

The RPP-WTP Project Manager shall ensure implementation of the Project Management Plan, thus assuring that adequate resources with appropriate technical background are available and organized to perform subsequent tasks. This activity also assures that the input information required for the safety standards and requirements identification process has been collected and organized. This input information includes the top-level safety standards and principles stipulated by DOE in DOE/RL-96-0006 and the laws and regulations applicable to the RPP-WTP project.

The DOE/RL-96-0004 safety requirements and standards identification Process Manager for the project is the Radiological, Nuclear, and Process Safety Manager.

The Process Manager chairs the DOE/RL-96-0004 safety requirements and standards identification Process Management Team (PMT). The PMT is constituted in accordance with project implementing documents and includes managers from the following project organizations:

- Environmental, Safety, & Health
- Engineering
- Operations

The Process Management Team shall oversee the ISM process and shall provide resources and resolve issues as necessary. The PMT shall set up integrated teams for the conduct of ISM on a plant system basis. Individual PMT members shall provide various subject matter experts to help fulfill the roles required of the Integrated Teams for conduct of the ISM process.

3.0 Identification of Work

The aim of this activity is to describe the work that will be performed so that the hazards inherent in the work can be identified and evaluated. Work activity experts who have extensive knowledge of the overall processing approach and are integrally associated with the facility design shall perform this activity. work activity experts shall be drawn from the following RPP-WTP organizations:

- Engineering staff
- Operations staff

When appropriate, the PMT may also draw work activity experts from the staff of other departments, such as from Construction.

In an overall sense, identification of work involves definition of the project mission and identification of the processes that must be performed to accomplish the mission. It includes selection of optimum functions, processes, and parameters through trade studies and definition of functional requirements. Identification of work for the purpose of design development involves definition of various plant systems, structures, and components. This latter definition is the focus for the Integrated Teams created to conduct ISM on a plant system basis.

The product of this activity includes:

- Process description
- System descriptions
- Descriptions of key structures
- Basis of design documents
- PFDs, MFDs, and P&IDs

The results of the identification of work activity shall be documented in the SRD by inclusion or by reference.

The identification of work activity is an iterative process. Identification of work will be reconsidered in light of design evolution, the outcome of hazard evaluations, and the development of hazard control strategies.

4.0 Hazard Evaluation

The aim of the hazard evaluation activity is to identify and characterize the hazards resulting from the work. The integrated teams shall conduct the hazard evaluation activity on a plant system basis. These teams shall include work activity experts (as defined in Section 3.0), hazard assessment experts, and hazard control experts.

Hazard assessment experts and hazard control experts shall generally be members of the technical staffs of the Safety Analysis Manager and of the Regulatory Safety Manager. The process management team shall provide additional technical resources as required to evaluate the hazards.

The hazard evaluation shall address hazards inherent in normal operation as well as potential accidents resulting from abnormal internal and external events.

The hazard evaluation shall comprise the following elements:

- Identification of Hazards
- Identification of Potential Accident/Event Sequences
- Estimation of Accident Consequences
- Estimation of Accident Frequencies
- Consideration of Common Cause and Common Mode Failures
- Definition of Design Basis Events
- Definition of Operating Environment
- Identification of Potential Control Strategies
- Documentation

These elements are discussed below.

4.1 Identification of Hazards

The objective of this element is to systematically identify the hazards associated with the defined work.

The integrated teams shall compile a list of hazardous materials and energy sources associated with the facility processes, design, and operations. This list shall be compiled based on the identified work. This compilation provides information used to identify potential accidents resulting in the uncontrolled release of hazardous material or energy to workers, the public, and the environment. The team may use checklists to guide the compilation process and to assure that all potential hazards from both natural and manmade sources originating from outside and inside the facility are addressed.

4.2 Identification of Potential Accident/Event Sequences

The objective of this element is to perform a structured and systematic examination of the facility and its operations to identify potential accidents (including common mode and common cause failures). The team shall conduct this examination using methodologies and guidelines in AIChE (1992).

4.3 Estimation of Consequences

4.3.1 Accident Severity Level Identification

A severity level, SL, shall be assigned to each postulated radiological accident. The severity level shall reflect the unmitigated consequences of the postulated accident. Unmitigated consequences shall account for the quantity, form and location of the radioactive material available for release, and the energy sources available to interact with the hazardous material. Unmitigated consequences shall not account SSCs that serve to prevent or mitigate the release. Specifically, unmitigated consequences shall be evaluated on the basis of a ground level release. The severity level shall be defined as follows:

SL	Facility Worker Consequence	Collocated Worker Consequence	Public Consequence
SL-1	> 25 rem/event	> 25 rem/event	> 5 rem/event
SL-2	5 - 25 rem/event	5 - 25 rem/event	1 - 5 rem/event
SL-3	1 - 5 rem/event	1 - 5 rem/event	0.1 - 1 rem/event
SL-4	< 1 rem/event	< 1 rem/event	< 0.1 rem/event

These severity levels are related to the radiological and process standards of SRD Chapter 2.0 as follows:

- The unmitigated consequences associated with SL-1 events exceed the radiological standards for extremely unlikely events (SRD Safety Criterion 2.0-1).
- The unmitigated consequences associated with SL-2 events are below the radiological standards for extremely unlikely events (SRD Safety Criterion 2.0-1).
- The unmitigated consequences associated with SL-3 events are below the radiological standards for unlikely events (SRD Safety Criterion 2.0-1).
- The unmitigated consequences associated with SL-4 events are below the radiological standards for anticipated events (SRD Safety Criterion 2.0-1).

Consequences to the facility worker shall be evaluated at the worst-case occupied location. Consequences to the collocated worker and the public shall be evaluated at the locations specified in Appendix D to the *Safety Requirements Document, Volume II*.

Early in the design, the severity level is estimated based on the experience of the Integrated Teams. As the design progresses, these estimates are confirmed through the formal accident analyses described in Section 4.3.2. These accident analyses do not address all of the potential accidents identified, but they do address bounding examples of each type of accident. The team should use the results of the accident analyses to validate the severity level estimates for potential accidents not addressed in the formal accident analyses.

The potential consequences of releases of hazardous chemicals shall also be assessed. The assessment shall consider both the inherent hazard of the chemical itself, and the potential for the chemical hazard to initiate or exacerbate a radiological hazard.

4.3.2 Accident Analysis

Accident analyses provide confirmation that the design satisfies the radiological and process standards in the SRD. Accident analyses also provide confirmation of the severity levels assigned to potential accidents.

The formal accident analyses shall address design basis external events and natural phenomena as well as postulated internal events.

The postulated internal events shall be grouped by type. Accident types applicable to the RPP-WTP include the following:

- Liquid spills
- Spills of solid materials
- Pressurized releases
- Chemical reactions
- Boiling
- Flammable gas ignition (e.g., hydrogen in air)
- Fires
- Load drops
- Radiation exposure
- Criticality

As a minimum, the accident analysis shall address the most severe credible event of each type.

Initially, the accident analysis shall evaluate the unmitigated consequences of the postulated accidents. As control strategies are developed, the accident analysis shall also evaluate the impact of the SSCs that implement the control strategy on the potential consequences.

The accident analysis shall consider the following factors:

- Inventory of material at risk in the scenario.
- The respirable release fraction for the accident scenario. This is a function of the composition of the material at risk, of the form of the material, and of the interaction between the material at risk and the energy available in the accident scenario.
- The fraction of the airborne material released to potentially occupied locations or the environment.
- Bounding atmospheric dispersion coefficients (if appropriate).
- Radiological composition of the material released.
- External radiation field.
- Exposure times.

The accident analysis shall address the potential consequence to facility workers, collocated workers, and the public.

4.3.3 Normal Conditions

Some hazards inherent in normal operation must be mitigated to comply with the standards for normal operation in SRD Chapter 2.0. Such hazards shall be addressed in accordance with the RPP-WTP Radiation Protection Plan.

4.4 Estimation of Accident Frequencies

There is normally insufficient information early in the design to accurately quantify the frequency of postulated internal events because this frequency depends on the design of the SSCs that implement the control strategy used to manage the hazard. At an early stage, frequency evaluations may be based on the team's experience with similar hazards in similar facilities. The team shall validate these estimates as the design develops.

As the design matures, information on the frequency of hazardous events is gained from the use of hazard evaluation techniques that provide frequency data (i.e., HAZOP, FMEA, Event Trees, and Fault Trees). Evaluations of the frequency of failure in redundant systems or in diverse systems using similar equipment shall consider dependent failures.

The frequencies of design basis external events may be derived from existing analyses (e.g., safety analyses for adjacent facilities), from evaluation of historical data (e.g., transportation data), or from site-specific information (e.g., seismic history).

4.5 Consideration of Common Cause/Common Mode failures

The following are typical common cause events:

- Natural phenomena events
- External man made events
- Loss of electrical power
- Fire
- Internal missiles
- Internal flooding

Common cause events should be treated as discrete events in the hazard analysis. The analyses of common cause events shall focus on identifying provisions to prevent the loss of safety function. The analyses of natural phenomena events shall consider induced effects, such as fire and loss of electrical power.

Common mode failures shall be addressed through dependent failure modeling as required by Section 4.4 above.

4.6 Definition of Design Basis Events

The hazard evaluation shall identify a set of internal design basis events. These events shall be selected to define a set of bounding performance requirements for the SSCs relied upon to control the hazards.

The hazard evaluation shall define a set of external man made design basis events. These events shall be selected based on the results of the hazard analysis to define a set of bounding performance requirements for the SSCs relied upon to mitigate these events.

The integrated teams perform the identification of internal and external design basis events.

Design basis natural phenomena shall be as defined in the SRD Safety Criteria 4.1-3 and 4.1-4.

4.7 Definition of Operating Environment

The hazard evaluation shall define a set of bounding operating conditions in which SSCs relied upon to control hazards must function. Environmental parameters to be addressed include the following:

- Temperature
- Pressure
- Humidity
- Radiation Levels
- Chemical Environment

4.8 Identification of Potential Controls

Based on the experience and judgement of team members, the integrated team shall identify an initial set of potential hazard controls to manage each potential accident. This set of potential hazard controls shall address means of preventing the potential accident and should address means of mitigating the consequences of the accident. The function(s) of each potential hazard control should be clearly described. Potential hazard controls shall be identified to manage accident conditions arising from upsets in the process, conditions arising from external events, and conditions inherent in the normal operation of the process.

4.9 Documentation

The hazard evaluation shall be documented in a hazard analysis report (HAR). The results of the hazard evaluation shall be contained in a hazard database. For each hazard considered, the hazard database shall record the following information produced by the hazard evaluation:

- Hazard identifier
- Hazard description
- Initiators
- Hazard severity level estimate (based on unmitigated consequences)
- Severity level basis
- Assumptions affecting the release (material at risk, energy available, etc)
- Hazard frequency estimate
- Basis for frequency estimate
- Potential controls and functions
- References for the hazard (these would typically be products of the work identification process)

Hazard evaluation documentation shall be included in the SRD by inclusion or by reference. This documentation shall include the following:

- Description of the comprehensive approach to hazard evaluation
- Description of the methodology for identification and quantification of work hazards
- Description of the methodology for identifying potential accident scenarios
- Description of the methodology for consequence assessment
- Clear identification of assumptions (e.g., quantity and form of material at risk, rate of release and relevant process conditions) that may drive or inhibit the potential accident must be clearly identified
- Description of results
- Evidence of appropriate staffing, and adequate technical staffing and structure applied to the hazard evaluation

5.0 Development of Control Strategies

The aim of the development of control strategies activity is to identify a means of controlling each of the hazards identified in the hazard evaluation. The integrated teams of work activity experts, hazard assessment experts, and hazard control experts, as discussed in Sections 3.0 and 4.0, perform this activity.

The PMT members shall provide additional technical resources as required to develop the control strategies.

The integrated teams select preferred control strategies based on the set of potential controls identified by the hazard evaluation team. Selection of the preferred strategy considers the following factors:

- The functions required of the strategy in order to control the hazard
- The degree of defense in depth and reliability provided by the control strategy. The Implementing Standard for Defense in Depth provides guidance in this area.
- Applicable design basis events.
- The operating environment in which the SSCs implementing the control strategy must function.
- Effectiveness and efficiency of the control strategy.
- Conformance with the DOE stipulated top level standards.
- Compliance with applicable laws and regulations.

The control strategy will typically comprise a series of elements including some or all of the following:

- Passive and/or active SSCs that function to prevent the release (that is, SSCs that reduce the probability that a release will occur)
- Passive and/or active SSCs that function to mitigate the release (that is, SSCs that reduce the consequences once a release has occurred)
- Administrative controls (for example, limits on inventory)

Consistent with the defense in depth principle, the control strategy development should emphasize preventive measures. It should also emphasize passive SSCs over active SSCs and retention of released material over dispersion. Ideally, the preferred control strategy should incorporate SSCs that prevent releases and SSCs that mitigate the consequences of a release, should it occur.

Once the preferred control strategy is identified, it shall be evaluated using the techniques described in Section 4.3 through 4.5. In addition, the evaluation of the control strategy shall identify the measures necessary to assure that it performs its functions reliably. Such measures include maintenance requirements, testing intervals and calibration frequency. The results of this evaluation serve to confirm that the control strategy is capable of satisfying SRD Safety Criteria 2.0-1.

If credit is taken for operator action to satisfy the public radiological exposure standards of Safety Criterion 2.0-1, adequate radiation protection is provided to permit access and occupancy of the control room or other control locations under accident conditions without personnel receiving radiation doses in excess of 5 rem TEDE whole body gamma and 30 rem beta skin for the duration of the accident. If credit is taken for operator action to satisfy public chemical exposure to the standards of Safety Criterion 2.0-2, provisions for operational access and control are made so that the operator exposure does not exceed the limits specified in Safety Criterion 4.3-7.

Documentation of the hazard control strategy development process shall clearly indicate selection of the control strategies and show the linkage of the control strategies to the respective hazards. The control strategy should be described in terms of the safety functions required (e.g., limit release of radionuclides, etc.) and in terms of a set of engineered features, administrative controls (procedures and training), and management systems selected for implementing the strategy. When the nature of the hazard is such that the appropriate control strategy is self-evident, the documentation need only demonstrate that the control strategy meets most, if not all, of the selection criteria, and need not provide a discussion of other, nonapplicable control strategies. Similarly, where a proven control strategy that is appropriate to the hazard exists and it is obvious to the team that there are no other alternative control strategies that could be equally attractive, then the documentation need only demonstrate that the control strategy meets most, if not all, of the selection criteria. Otherwise, the documentation should identify all control strategies considered and provide a defensible rationale for selection of the preferred strategy.

The following information produced by the control strategy definition shall be recorded in the hazard database:

- Preferred control strategy
- Linkage of the control strategy to the respective hazards
- Rationale for preferred control strategy selection
- Defense in depth provided
- Control strategy functions and performance requirements
- Estimate of the unmitigated event frequency
- Estimate of the consequences from the mitigated event
- Estimate of the mitigated event frequency
- Applicable design basis events (e.g., design basis earthquake)

One of the issues in developing a control strategy for a particular hazard is determining the number of layers of prevention and mitigation appropriate for the hazard. The control strategies shall conform to the requirements defined in the Implementing Standard for Defense in Depth. In addition, the following guidance shall be considered in developing control strategies.

The general RPP-WTP design approach is to provide two confinement barriers against the release of hazardous materials. The process vessels and piping form the primary confinement barrier; the process cells and associated ventilation system form the secondary confinement barrier. Releases from the primary confinement are mitigated by the secondary confinement.

The accident severity levels defined in Section 4.3.1 are related to the exposure standards in SRD Safety Criterion 2.0-1. The SRD Safety Criterion 2.0-1 exposure standards are frequency based, so it is possible to establish target frequencies for events with a given severity level. The target frequencies tabulated below are consistent with SRD Safety Criterion 2.0-1.

SL	Event Target Frequency (yr^{-1})
SL-1	$<10^{-6}$
SL-2	$<10^{-4}$
SL-3	$<10^{-2}$
SL-4	$<10^{-1}$

These target frequencies may be used to guide control strategy development as described below.
For SL-1 events:

- Meeting the target frequency will usually require a control strategy that incorporates diverse and independent SSCs that act to prevent and mitigate the event.
- Meeting the target frequency will usually require diverse SSCs that act to prevent the release.
- The degree of mitigation required depends on the release frequency, that is, on the reliability of the preventive SSCs. For example, assume that the preventive SSCs assure that the frequency of release is less than 10^{-4} per year, but more than 10^{-6} per year. This frequency is not acceptable for events that have SL-1 level consequences, but is acceptable for events that have SL-2 level consequences. Therefore, the control strategy would need to provide enough mitigation to reduce the consequences of the release to the levels associated with a SL-2 event, as a minimum. The combined reliability of the preventive SSCs and the SSCs that provide mitigation needs to satisfy the target frequency for a SL-1 event. That is, the probability that the SSCs that provide mitigation will fail should be on the order of 10^{-2}, given the release.
- SSCs in control strategies for SL-1 events shall satisfy the single failure criteria in the Implementing Standard for Defense in Depth.

For SL-2 events:

- Meeting the target frequency will usually require a control strategy that incorporates diverse and independent SSCs that act to prevent and mitigate the event.
- The degree of mitigation required depends on the release frequency, that is, on the reliability of the preventive SSCs. For example, assume that the only viable preventive SSCs assure that the frequency of release is less than 10^{-2} per year, but more than 10^{-4} per year. This frequency is not acceptable for events that have SL-2 level consequences, but is acceptable for events that have SL-3 level consequences. Therefore, the control strategy would need to provide enough mitigation to reduce the consequences of the release to the levels associated with a SL-3 event, as a minimum. The combined reliability of the preventive SSCs and the SSCs that provide mitigation needs to satisfy the target frequency for a SL-2 event. That is, the probability that the SSCs that provide mitigation will fail should be on the order of 10^{-2}, given the release.
- SSCs in control strategies for SL-2 events should satisfy the single failure criteria in the Implementing Standard for Defense in Depth.

For SL-3 and SL-4 events:

- The mitigation provided by the secondary confinement would be adequate to satisfy SRD Safety Criterion 2.0-1. It would also be adequate to satisfy SRD Safety Criteria 1.0-3 through 1.0-5. However, preventive features should be considered consistent with the defense in depth principle.
- A single preventive SSC may satisfy the frequency goal for SL-3 and SL-4 events.
- SSCs in control strategies for SL-3 and SL-4 events need not satisfy the single failure criteria in the Implementing Standard for Defense in Depth.

Notwithstanding the foregoing guidance on control strategy selection, administrative controls alone may be credited as the controls that protect facility workers, when appropriate. Timely evacuation from the vicinity of the hazard is considered to be an administrative control.

6.0 Classification of Structures, Systems, and Components

The design classification process used on the RPP-WTP Project provides a consistent, project-wide approach for the classification of the RPP-WTP SSCs based on their importance to controlling normal releases and accident prevention and mitigation. This approach ensures that SSCs are designed, constructed, fabricated, installed, tested, operated, and maintained to quality standards commensurate with the importance of the functions that need to be performed. As the facility moves to deactivation, and the safety functions change, the classification of SSCs can be revised as necessary.

The RPP-WTP project has established a design classification system to provide assurance to DOE that the defined safety functions of SSCs will perform as intended.

SSCs defined as Important-to-Safety for the RPP-WTP include the following:

1) SSCs needed to prevent or mitigate accidents that could exceed public or worker radiological and chemical exposure standards of Safety Criteria 2.0-1 and 2.0-2 and SSCs needed to prevent criticality. This set of SSCs includes both the front line and support systems needed to meet these exposure standards or to prevent criticality. This set of Important-to-Safety SSCs are designated as Safety Design Class, as defined by SRD Safety Criterion 1.0-8.
2) SSCs needed to achieve compliance with the radiological or chemical exposure standards for the public and workers during normal operation; and SSCs that place frequent demands on, or adversely affect the function of, Safety Design Class SSCs if they fail or malfunction. This set of Important-to-Safety SSCs are designated as Safety Design Significant, as defined by SRD Safety Criterion 1.0-8.

   The processes for identifying the SSCs for each of the two groups of SSCs Important-to-Safety and the requirements assigned to each of the two groups are discussed below.

   Safety Design Class SSCs typically are identified by the results of accident analyses that show the potential for exposure standards to be exceeded or prevent a criticality. However, additional items may also be designated Safety Design Class independent of a specific accident analysis. These are items that protect the facility worker from potentially serious events. Typically, these events are deemed to present a challenge to the facility worker severe enough that mitigation is prudent, without the need to perform a specific consequence analysis.

   Safety Design Significant SSCs are identified in several ways including: (1) SSCs identified as significant contributors to safety by the analyses that confirm the facility accident risk goals are met (this is one way to identify SSCs that place frequent demands on, or adversely affect the function of, Safety Design Class SSCs if they fail or malfunction), (2) SSCs that are needed to ensure that standards for normal operation are not exceeded (e.g., bulk shield walls or radiation monitors), (3) SSCs selected based on the dictates of nuclear and chemical facility experience and prudent engineering practices, and (4) SSCs whose failure could prevent Safety Design Class SSCs from performing their safety function (e.g., Seismic II/I items).

When an SSC is designated as Safety Design Class it has the following attributes:

1) Quality Level 1 (QL-1) is applied to the SSC to provide added assurance that the SSCs can perform their specified safety function.
2) For an active system or component, the safety function is preserved by application of defense-in-depth such that failure of the system or component will not result in exceeding a public or worker accident exposure standard. For a mitigating feature, this means that, given that the accident has occurred, the consequence of the accident will not result in exceeding a public or worker exposure standard. For a preventative feature, this means that the failure of the system or component will not allow the accident to occur and progress such that a public or worker accident exposure standard is exceeded. If the hazard analysis shows that these requirements are necessary, this requirement may be achieved by designing the Safety Design Class system or component to withstand a single active failure or by designating two separate and independent systems or components as Safety Design Class.
3) The SSC is designed to withstand the effects of natural phenomena such that it can perform any safety functions required as a result of a natural phenomena event in accordance with Safety Criterion 4.1-3.
4) General design requirements are applied as identified in Chapter 4.0 of the SRD for Safety Design Class SSCs.
5) Specific design requirements based on the type of component are applied as invoked in SRD Chapter 4.0.
6) Other design requirements may be applied based on the specific safety function to be performed by the Safety Design Class SSC. This specific safety function is determined from the accident analysis that identified the need for prevention or mitigation by Safety Design Class SSCs.
7) Operational requirements (e.g., periodic testing and preventative maintenance) are applied to Safety Design Class SSCs through the application of Technical Safety Requirements.

When an SSC is classified as Safety Design Significant it is has the following attributes.

1) Quality Level 2 (QL-2) is applied to the SSC to provide added assurance that the SSCs can perform their specified safety function.
2) The SSC is designed to withstand the effects of natural phenomena such that it can perform its safety functions required as a result of a natural phenomena event in accordance with Safety Criterion 4.1-4.
3) General and specific design requirements are applied as identified in Chapter 4.0 of the SRD for Safety Design Significant SSCs.
4) Other design requirements again may be applied based on the specific safety function to be performed by the Safety Design Significant SSC.

7.0 Identification of Standards

Identification of standards is an iterative activity. Initially, the set of standards and requirements is derived from a general understanding of the hazards inherent in the work. As the design evolves, the hazard evaluation and the development of the control strategies justify tailoring the set of standards to better fit the hazards.

The identification of engineering/design, manufacture/fabrication, and construction standards is performed by an integrated team including work activity experts, hazard assessment experts, hazard control experts, as discussed in Sections 3.0 and 4.0, and standards experts. Identification of other standards (e.g., quality assurance, conduct of operations, etc.) will be performed by specially constituted teams formed by the PMT. The aim of this activity is to identify a tailored set of standards and requirements that will assure adequate safety when implemented.

The process management team shall provide additional technical resources as required to identify the standards.

Standards experts shall be drawn from the following RPP-WTP organizations:

- Staff of the Engineering Manager
- Technical staff of the ES&H Manager

The standards identified are evaluated and tailored for each control strategy based on compliance with applicable laws and regulations and conformance with the DOE-stipulated top level standards, plus the output of the preceding hazard evaluation and control strategy development steps. Typical considerations include the following:

- The severity level of the hazard
- The number of independent SSCs that comprise the control strategy
- The control strategy functions - recognizing that a specific control strategy may have multiple functions and serve to control multiple hazards
- The service environment
- The applicable design basis events
- The target reliability for the control strategy

The target frequencies described in Section 4 provide a basis for establishing target reliabilities for the SSCs that comprise the control strategy. The combined reliability of the preventive SSCs and the SSCs that provide mitigation must be consistent with the target frequency for the unmitigated event. The reliability of the preventive SSCs should be consistent with the release frequency used to determine the degree of mitigation provided.

Documentation of the standards and requirements identification process provides justification of the set selected and links each control strategy to its associated set of standards. The information generated during standards selection is retained in database form for each control strategy:

- Control strategy
- Service environment
- Applicable design basis events
- Applicable standards
- Performance requirements
- Testing/calibration requirements
- In-service inspection requirements
- Maintenance requirements
- Quality level
- Standards justification

This information is structured so it can be linked to the control strategies in the hazard schedule. This provides a link from the hazards through the control strategies to the standards. Not all of this information will be available early in the design. For example, it will not be possible to define maintenance and testing requirements until the design is mature.

The standards identified through this activity shall be reflected in the SRD.

As the standards are tailored, discrepancies with the current version of the SRD may arise. Such discrepancies shall be recorded. Formal changes to the SRD require approval from DOE.

8.0 Confirmation of Standards

Based on the recommendation of the PMT, the RPP-WTP Project Safety Committee (PSC) Chair requests the PSC to confirm the selected set of standards. The PSC defines a review approach, carries out the review, and documents the findings of the review. Comments by the PSC shall receive formal disposition by the Process Management Team.

9.0 Formal Documentation

Following confirmation by the PSC, the results of the standards selection process shall be documented in the Safety Requirements Document (SRD). The SRD shall incorporate documentation supporting these results by reference. The SRD shall identify and justify the set of requirements and standards selected to provide adequate protection of workers, the public, and the environment.

10.0 Recommendation

The recommended set of standards shall be certified in accordance with project implementing documents. When properly implemented, the set of standards:

1) Provides adequate safety
2) Complies with applicable laws and regulations
3) Conforms with the Top-Level Safety Standards and Principles

11.0 Definitions

Credible event: Any event with a frequency greater than 10^{-6} per year, including allowance for uncertainties.

Important to Safety: Structures, systems, and components that serve to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the workers and the public. It encompasses the broad class of facility features addressed (not necessarily explicitly) in the top-level radiological, nuclear, and process safety standards and principles that contribute to the safe operation and protection of workers and the public during all phases and aspects of facility operations (i.e., normal operation as well as accident mitigation).

This definition includes not only those structures, systems, and components that perform safety functions and traditionally have been classified as safety class, safety-related, or safety-grade, but also those that place frequent demands on or adversely affect the performance of safety functions if they fail or malfunction, i.e., support systems, subsystems, or components. Thus, these latter structures, systems, and components would be subject to applicable top-level radiological, nuclear, and process safety standards and principles to a degree commensurate with their contribution to risk. In applying this definition, it is recognized that during the early stages of the design effort all significant systems interactions may not be identified and only the traditional interpretation of important to safety, i.e., safety-related, may be practical. However, as the design matures and results from risk assessments identify vulnerabilities resulting from non-safety-related equipment, additional structures, systems, and components should be considered for inclusion within this definition.

Mitigated event: As used in this standard, a mitigated event involves the following sequence:

- An initiating event that could lead to a release from the primary confinement barrier
- Failure of all elements of the control strategy that would prevent the initiating event from developing into a release from the primary confinement barrier
- Mitigation of the consequences of the release as provided by the control strategy

Mitigated event frequency: The mitigated event frequency is the corresponding release frequency times the probability that the elements of the control strategy that mitigate the release will function given the release.

Release frequency: The release frequency is the product of the frequency of the initiating event times the probability that all elements of the control strategy that would prevent the release fail, given the initiating event.

Reliability: The probability that an SSC will perform its safety function when required.

Unmitigated event: As used in this standard, an unmitigated event involves the following sequence:

- An initiating event that could lead to a release from the primary confinement barrier
- Failure of all elements of the control strategy that would prevent the initiating event from developing into a release from the primary confinement barrier
- Failure of all elements of the control strategy that would mitigate the consequences of the release

Unmitigated event frequency: The frequency of an unmitigated event is the corresponding release frequency times the probability that all elements of the control strategy that would mitigate the release fail, given the release.

Appendix B

Implementing Standard for Defense in Depth

CONTENTS

1.0 Introduction

The purpose of this Implementing Standard is to consolidate the standards to be applied in the design, construction, and operation of the RPP-WTP with respect to defense in depth. This Implementing Standard also provides for tailoring of defense in depth as is appropriate to the nature and severity of the hazard and hazardous situations to which it is applied.

Section 2.0 identifies the subordinate standards used in the application of the six defense in depth sub-principles of DOE/RL-96-0006. These subordinate standards are derived, in part, from various available consensus standards. In cases where no relevant consensus standard exists for a given defense in depth sub-principle, this document provides the criteria to be implemented.

Section 3.0 discusses the approach to be used in implementing defense in depth with respect to determining an adequate combination of passive barriers and active SSCs that afford protection against a postulated initiating event.

Terms used in this Implementing Standard are defined in Section 4.0. These definitions are derived from consensus standards, tailored to the work and hazards of the RPP-WTP.

2.0 Standards for the Implementation of Defense in Depth Sub-Principles

The Top Level Principles identify the following sub-principles that must be addressed in order to demonstrate compliance with the principle of defense in depth:

- Defense in depth
- Prevention
- Control
- Mitigation
- Automatic Systems
- Human Aspects

The following subsections contain the standards on application of the six sub-principles of defense in depth from DOE/RL-96-0006 (Ref. 5.4). These consensus standards will be tailored to remove obviously reactor-specific and other non-applicable criteria. In accordance with the DOE/RL-96-0004 process, further tailoring will be performed as the design develops.

The following subsections contain excerpts and extracts from several consensus standards. Where necessary to avoid the implication of misquoting, differences in wording from the cited consensus standards are identified by presenting added words in italics and by inserting double-brackets where words have been removed. Citation of a portion of a given consensus standard shall not be read to infer that other portions of the standard not specifically cited are being invoked.

2.1 Defense in Depth

"To compensate for potential human and mechanical failures, a defense-in-depth strategy should be applied to the facility commensurate with the hazards such that assured safety is vested in multiple, independent safety provisions, not one of which is to be relied upon excessively to protect the public, the workers or the environment. This strategy should be applied to the design and operation of the facility." (DOE/RL-96-0006, Section 4.1.1.1)

2.1.1 Implementing Standards

1. DOE O 420.1 (Ref. 5.2), Section 4.1.1.2, first three paragraphs only
2. Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosives Safety Criteria (Ref. 5.3), Section 2.3, except last paragraph
3. ANSI/ANS-58.9-1981, Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems (Ref. 5.8)
4. IEEE Std 379-1994, IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems (Ref. 5.9)

2.1.2 Discussion

The RPP-WTP will be designed with the objective of providing multiple *levels* of protection to prevent or mitigate the unintended release of radioactive materials to the environment. Defense in depth will include: siting; minimization of material at risk; the use of conservative design margins and quality assurance; the use of successive physical barriers for protection against the release of radioactivity; the provision of multiple means to control critical safety functions (those basic safety functions needed to control the processes, maintain them in a safe state, and to confine and mitigate radioactivity associated with the potential for accidents with significant [ ] radiological impact *to the public, facility workers or collocated workers*); the use of equipment and administrative controls which restrict deviations from normal operations and provide for recovery from accidents to achieve a safe condition; means to monitor accident releases required for emergency responses; and the provision of emergency *preparedness* for minimizing the effects of an accident (Ref. 5.2).

The defense-in-depth concept is integrated into the RPP-WTP design process. The application of the defense-in-depth concept to the facility design helps identify potential safety features to be included in the facility design. Consideration will be given to prevent or mitigate accident consequences from contaminating the environment, even when direct public or worker safety is not an issue.

Defense in depth is a safety design concept or strategy that is applied at the beginning and will be maintained throughout the facility design process. This safety design strategy is based on the premise that no one *level* of protection is completely relied upon to ensure safe operation. This safety strategy provides multiple *levels* of protection to prevent or mitigate an unintended release of radioactive material to the environment.

Conceptually, there are three levels of defense in depth.

1. The first level of defense consists of a well-designed facility with process design to reduce source terms, reliable SSCs that are simple to operate and maintain and resistant to degradation, and personnel well trained in operations and maintenance and committed to a strong safety culture.
2. The second level recognizes that failures of systems and components and human failures cannot be entirely eliminated and that protective features (e.g., engineering design features and administrative controls) are required. These features are provided to ensure a return to normal operation or to bring the facility to a safe condition in the event of anticipated, but abnormal events. These features may provide automatic system response to such events or may be monitors that alert operators to the necessity of taking manual action. Such response to off-normal conditions can effectively halt the progression of events toward an accident.
3. The final level of defense consists of conservatively designed *important to* safety SSCs to prevent or mitigate the consequences of accidents that may be caused by errors, malfunctions, or events that occur both internal and external to the facility (Ref. 5.3).

Implementing Standards for the following elements of defense in depth described in the nonreactor safety Implementation Guide (IG) related to safety design and construction are addressed in the sections of this document that are referenced below.

IG Element	Discussed in Section
Siting	2.2.2
Material at risk	2.2.2
Conservative design	2.2.2
Quality assurance	2.6.2
Physical barriers	2.4.2
Critical safety functions	2.3.2
Equipment and administrative controls	2.3.2 and 2.6.1
Emergency features	2.5.2

When active SSCs are required to achieve defense in depth, RPP-WTP will apply the single failure criterion in accordance with ANSI/ANS-58.9 (Ref. 5.8) for fluid systems and IEEE Std 379 (Ref. 5.9) for electrical and instrumentation and control systems, as discussed below.

The application of the single failure criterion begins with the identification of an initiating event. Initiating events are identified in the normal course of applying integrated safety management in accordance with DOE/RL-96-0004, as described in the RPP-WTP Implementing Standard for Safety Standards and Requirements Identification (i.e., SRD Vol. II, Appendix A). In evaluating the defense in depth of the RPP-WTP, single failures must be postulated in addition to the initiating event (that is the initiating event is not the single failure) (Ref. 5.8). For fluid systems, during the short term, the single failure considered may be limited to an active failure. During the long term, assuming no prior failure during the short term, the limiting single failure considered can be either active or passive. Examples of passive failures are valve packing and pump seal leakage.

Tailoring of the application of the single failure criterion to the work and associated hazards is discussed in Section 3.0.

2.2 Prevention

"Principal emphasis should be placed on the primary means of achieving safety, which is the prevention of accidents, particularly any that could cause an unacceptable release." (DOE/RL-96-0006, Section 4.1.1.2)

2.2.1 Implementing Standards

1. DOE O 420.1 (Ref. 5.2), Section 4.1.1.2, first three paragraphs only
2. Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosives Safety Criteria (Ref. 5.3), Section 2.3, except last paragraph

2.2.2 Discussion

The provision of hazard elimination and protection shall be optimized by measures such as the choice of siting, proven conservative design and construction, a robust start-up testing program, operating requirements (i.e., clear definition of normal and abnormal operating conditions and maintenance activities).

Siting. The RPP-WTP site location will reduce the need to provide design measures to alleviate potentially hazardous conditions or to protect surrounding populations (for example, consideration of ground instability, river flooding, and hazards due to nearby industrial installations or activities) (Ref. 5.3).

Material at Risk. The RPP-WTP and its process design and administrative controls will minimize and control inventories of radioactive materials and their forms (Ref. 5.3).

Conservative Design. The RPP-WTP design will include conservative margins that allow flexibility of operations and maximize the time before requiring corrective actions. These margins will also take into consideration the potential degradation of elements and operational errors (Ref. 5.3).

The design shall address all identified hazards and hazardous situations and pursue methods for their prevention. The preferred means of prevention is to eliminate or reduce the severity of the hazard itself. According to the Implementation Guide on nonreactor facility safety, one objective of prevention as an element of defense in depth is to apply facility and process design and administrative controls to minimize and control inventories of radioactive materials and their forms (that is, minimize the material at risk) (Ref. 5.3).

Elimination or reduction of the hazard can be achieved by substituting less hazardous materials in processing, limiting the inventory of the material, etc. The design process must provide evidence through documentation that this option was considered and implemented to the maximum extent practicable. Where the hazard itself cannot be eliminated or reduced, controls shall be provided to reduce the likelihood of the hazard manifesting itself into an accident. The criterion for acceptability is discussed in Section 3.0. Where hazard elimination is not practicable, passive features are to be employed, since they are simple and have a high degree of reliability. Where this is not practicable, active protection will be proposed that has a degree of reliability and confidence commensurate with the potential hazard severity.

To illustrate the differences between hazard elimination and the provision of passive or active protection, consider the need for a cask lift using a crane. Elimination of the hazardous situation (inherent safety) is removal of the potential for raising a cask above its safe drop height by ensuring that the building dimensions physically prevent a lift above that height at all points of travel. If this were not practical, the provision of a physical stop would be passive protection to prevent a lift above the safe drop height. Active protection systems would include limit switches and braking systems. Procedures and operator training would ensure that the crane is handled and operated in a way that maximizes safety (e.g., check security of load, minimum lift height to confirm security, no challenge to engineered systems, and exclusion of personnel from load lifting area).

2.3 Control

"Normal operation, including anticipated operational occurrences, maintenance and testing, should be controlled so that facility and system variables remain within their operating ranges and the frequency of demands placed on structures, systems and components important to safety is small." (DOE/RL-96-0006, Section 4.1.1.3)

2.3.1 Implementing Standards

1. DOE O 420.1 (Ref. 5.2), Section 4.1.1.2, first three paragraphs only
2. Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosives Safety Criteria (Ref. 5.3), Section 2.3, except last paragraph
3. IEEE Std 603-1991, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations (Ref. 5.11)
4. ISA-S84.01-1996, Application of Safety Instrumented Systems for the Process Industries (Ref. 5.13)

2.3.2 Discussion

The DOE Implementing Guidance for nonreactor facility safety provides two criteria related to the defense in depth sub-principle of control:

Critical safety functions. Design to provide multiple ways for safety functions to control processes, to maintain processes in a safe state, and to confine radioactivity when accidents could have the potential for significant [ ] radiological impact *to the public, facility workers or collocated workers* (Ref. 5.3).

Equipment and administrative controls. Include features to control process variables to values within safe conditions, to alert operating personnel of an approach toward conservative process limits, to allow timely detection of failure or malfunction of critical equipment, and to allow for the imposition of administrative controls assumed in the hazard analysis, and/or accident analysis (Ref. 5.3).

Normal operations, which include anticipated operational occurrences and maintenance and testing activities, shall be controlled so that facility and system parameters remain within their specified operating ranges and that the frequency of demands placed on SSCs for hazard prevention and mitigation is small.

This will be achieved by the choice of design that will:

1. Control key operating parameters such that facility operations remain within the safe operating envelope. Key operating parameters are those that define how the plant will be operated safely.
2. Maintain the safe operating envelope (e.g., a wide variation in operation conditions can be tolerated without entering into a potentially unsafe region).
3. Ensure that any failure mode would not move the facility or process toward a potentially unsafe region (i.e., fail to safe state).
4. Provide instrumentation and control features (e.g. temperature, pressure, radiation monitoring) which will warn of reduced margins of safety and, where appropriate, automatically return the process into the designated safe operating regime.
5. Achieve independence between SSCs credited for control of normal facility operations and those credited for prevention and mitigation of potential hazards.

For example, assume that the normal operating temperature range in an ion exchange column is set at 30 - 50 °C and that column temperatures above 80 °C lead to enhanced resin degradation and a potential explosion hazard. Engineered controls for maintaining that temperature within the normal operating limits (e.g., temperature control system) will be independent of that which would alert the operator and perform a preventative action (e.g., shut down process, increase cooling, etc.) in order that the hazard could not occur.

2.4 Mitigation

"The facility should be designed to retain the radioactive material through a conservatively designed confinement system for the entire range of events considered in the design basis. The confinement system should protect the workplace and the environment." (DOE/RL-96-0006, Section 4.1.1.4)

2.4.1 Implementing Standards

1. DOE O 420.1 (Ref. 5.2), Section 4.1.1.2, first three paragraphs only
2. Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosives Safety Criteria (Ref. 5.3), Section 2.3, except last paragraph
3. Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02, see below)

2.4.2 Discussion

The purpose of mitigation is to ensure reduction of consequences from potential hazards and hazardous situations. One method of achieving this element of defense in depth is to ensure that suitable confinement of radioactive and hazardous material is maintained throughout normal operation and credible accident conditions. Confinement will be achieved by physical barriers and by other SSCs that either assure integrity of the physical barriers or minimize the quantity and characteristics of any hazardous material potentially releasable.

DOE Order 420.1, Chg 2, requires:

> "All nuclear facilities with uncontained radioactive materials (as opposed to material contained within drums, grout and vitrified materials) shall have means to confine them. Such confinement will act to minimize the spread of radioactive materials and the release of radioactive materials in facility effluents during normal operations and potential accidents. For a specific nuclear facility, the number and arrangement of confinement barriers and their required characteristics shall be determined on a case-by-case basis. Factors that shall be considered in confinement system design shall include type, quantity, form, and conditions for dispersing the material. Engineering evaluations, trade-offs, and experience shall be used to develop practical designs that achieve confinement system objectives. The adequacy of confinement systems to effectively perform the required functions shall be documented and accepted through the Safety Analysis Report." (Ref. 5.2)

The DOE nonreactor facility safety Implementation Guide defines confinement barriers to include primary confinement and secondary confinement. "Primary confinement provides confinement of hazardous material to the vicinity of its processing -- typically by means of piping, tanks, glove boxes, encapsulating material, etc., along with any offgas systems that control effluent from the primary confinement. As such, primary confinement addresses the preventive sub-principle of defense in depth, as well as mitigation. Secondary confinement consists of a cell or enclosure surrounding the process material or equipment along with any associated ventilation exhaust systems from the enclosed area." [ ] (Ref. 5.3)

The RPP-WTP will provide physical barriers to confine radioactive material and thereby prevent uncontrolled releases. In general, multiple physical barriers - i.e., primary and secondary confinement - will be provided, especially for the most severe hazards and hazardous situations. Although RPP-WTP buildings will afford a tertiary confinement, as defined in the Implementation Guide, the RPP-WTP accident analysis will not take credit for holdup of radioactive materials by the buildings. The provision of multiple physical barriers will be tailored to the work and associated hazards, as discussed in Section 3.0.

The DOE nonreactor facility Implementation Guide (IG) suggests several industry consensus codes and standards for the design and construction of the SSCs comprising confinement, as follows: structures - IG subsection 5.2.1, ventilation systems - subsection 5.2.2.1, and process equipment - subsection 5.2.2.2. The specific standards for SSCs that implement mitigation with respect to SSCs comprising confinement are contained in the following Safety Criteria from the Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02):

- Structures - SC 4.1-2
- Ventilation systems - SC 4.4-6 through 4.4-8
- Process equipment - SC 4.2-1 through 4.2-3

2.5 Automatic Systems

"Automatic systems should be provided that would place and maintain the facility in a safe state and limit the potential spread of radioactive materials when operating conditions exceed predetermined safety setpoints." (DOE/RL-96-0006, Section 4.1.1.5)

2.5.1 Implementing Standards

1. IEEE Std 603-1991, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations (Ref. 5.11)
2. ISA-S84.01-1996, Application of Safety Instrumented Systems for the Process Industries (Ref. 5.13)
3. ANSI/ANS-58.8-1994, Time Response Design Criteria for Safety-Related Operator Actions (Ref. 5.7)

2.5.2 Discussion

Automatic systems shall be provided to prevent the facility from entering into or remaining within an unsafe regime that may lead to the potential for radioactive or hazardous material release to workers, the public, or the environment, except as discussed below. The definition of the boundaries between safe and unsafe regimes will be determined as a result of detailed facility design, start-up, and testing activities. This will allow the derivation of the predetermined setpoints for safe facility operations. Automatic systems will be part of the overall suite of SSCs provided as part of the hazard control strategy. The determination of the need for automatic systems will be assessed as part of the determination of the overall hazards control strategy.

Means shall be provided to automatically initiate and control all protective actions except as justified below. The design of important to safety systems shall be such that the operator is not required to take any action prior to the time described below following the onset of any event (Based on IEEE Std 603-1991 - Ref. 5.11).

Credit for operator action may be permissible only if safety analysis demonstrates that the total time interval required to perform the operator action exceeds the time at which the limiting design requirement would be reached without operator action, in accordance with the methodology of ANSI/ANS-58.8-1994 (Ref. 5.7).

2.6 Human Aspects

"The human aspects of defense in depth should include a design for human factors, a quality assurance program, administrative controls, internal safety reviews, operating limits (Technical Safety Requirements), worker qualification and training, and the establishment of a safety/quality program." (DOE/RL-96-0006, Section 4.1.1.6)

2.6.1 Implementing Standards

1. IEEE Std 1023-1988, IEEE Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations (Ref. 5.12).
2. Implementing standards for the quality assurance program, administrative controls, internal safety reviews, operating limits (Technical Safety Requirements), worker qualification and training, and the establishment of a safety/quality program are contained in the Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), as discussed below.

2.6.2 Discussion

<u>Design for Human Factors</u>

The design shall apply human factors engineering (HFE) to address the ergonomic requirements of facility operations and maintenance of the RPP-WTP. The DOE nonreactor Implementation Guide recommends that the following human factor elements be considered: equipment labeling, workplace environment (temperature and humidity, lighting, noise, vibration, and aesthetics), human dimensions, operating panels and controls, component arrangement, warning and annunciator systems, and communication systems (Ref. 5.3).

The RPP-WTP design engineers, in consultation with operators, will apply these HFE elements in the design of important to safety SSCs to ensure that operational preferences are implemented. Human factors engineering specialists will provide support in the application of HFE.

Human factors engineering shall be conducted in accordance with IEEE Std 1023-1988 (Ref. 5.12), as discussed below. Selection of this subordinate standard comports with the nonreactor safety Implementation Guide (Ref. 5.3).

IEEE Std 1023-1988 was developed specifically for nuclear power generating stations. Therefore, this subordinate standard will be tailored to the work and hazards of the RPP-WTP as follows. The formal HFE process described in subsection 6.1.1 of IEEE Std 1023-1988 will be applied to the evaluation of hazards whose consequences fall into the two highest severity levels - SL-1 and SL-2, with the following clarification:

> The project does not plan on constructing a separate plant simulator or physical mockup. The RPP-WTP distributed control system (DCS) - including the main control room panels -- is a programmable computer system. The project envisions having the DCS built, delivered to the site and proof-tested with the aid of the facility operators well in advance of plant startup. Therefore, a dynamic simulation capability for personnel training will be provided for SSCs with significant human interfaces that involve complex and interactive processes (Ref. IEEE Std 1023-1988 §§ 6.1.1.12 and 6.1.1.18).

Although the structured HFE program outlined in subsection 6.1.1 of IEEE Std 1023-1988 will not be implemented for SL-3 and SL-4 events, the general HFE elements will be considered for all ITS SSCs, as committed above.

Similarly, formal consideration of the HFE techniques and methodologies recommended in Section 5 of IEEE Std 1023-1988 will be undertaken for hazards of severity levels SL-1 and SL-2. Certain of these techniques and methodologies may be utilized in the evaluation of SL-3 and SL-4 events in the context of the normal design and hazard assessment and control effort, as part of the integrated safety management process.

<u>Quality Assurance Program</u>

The *Safety Requirements Document Volume II* (24590-WTP-SRD-ESH-01-001-02) Safety Criteria 1.0-10 and Section 7.3 require the RPP-WTP contractor to establish and implement a Quality Assurance Program compliant with 10 CFR 830.120. This program is being implemented in accordance with the *Quality Assurance Manual* (24590-WTP-QAM-QA-01-001).

The *Quality Assurance Manual* (24590-WTP-QAM-QA-01-001) applies specifically to work performed on or for the RPP-WTP. The QAP is in conformance with 10 CFR 830.120 (Ref. 5.1) and with the top-level principles stated in DOE/RL-96-0006 (Ref. 5.4).

<u>Administrative Controls</u>

Administrative controls include features to control process variables to values within safe conditions, to alert operating personnel of an approach toward conservative process limits, to allow timely detection of failure or malfunction of critical equipment, and to allow for the imposition of administrative controls assumed in the hazard analysis, and/or accident analysis (Ref. 5.3).

The primary means of implementing defense in depth is through the provision of multiple physical barriers that maintain confinement. The output of the design process, through which hazards and hazardous situations are identified, control strategies implemented and standards defined will be a set of SSCs that achieve defense in depth. SSCs so identified will always be backed up by administrative controls such as procedures. Administrative controls that afford a measure of defense in depth will be developed prior to facility operations. For the purpose of protecting the public and collocated worker, administrative controls alone shall not be relied on for the implementation of defense in depth. Administrative controls alone may be credited as the controls that protect facility workers, when appropriate. In such cases, defense in depth is provided through other human aspects, such as worker qualification and training.

Internal Safety Reviews

The Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), Safety Criterion 7.1-3, requires that the RPP-WTP contractor establish a safety framework and specifies requirements for the Internal Safety Oversight program consistent with Top-Level Principle 4.4.1, "Safety Review Organization". BNI has established a RPP-WTP Project Safety Committee (PSC) to provide an independent, interdisciplinary evaluation of matters related to nuclear, radiological, and process safety.

Operating Limits (Technical Safety Requirements)

The Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), Safety Criterion 9.2-1, commits the RPP-WTP contractor to prepare, submit for approval, and operate the facility in accordance with Technical Safety Requirements (TSRs). SCs 9.2-2 through 9.2-6 provide the safety criteria for the bases and contents, updating, submission for regulatory approval, and maintenance of TSRs.

As part of hazard evaluation, the role of the operator in the development of a potential hazard will be identified and reliability assessed. Human factors specialists in the multidisciplinary team will support this evaluation. The results of the assessment will be incorporated into administrative controls such as operating procedures and TSRs.

Worker Qualification and Training

The Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), Section 7.2, commits the RPP-WTP contractor to establish and implement a training program. Consistent with Top-Level Principles 4.3.4.1, "Personnel Training", 4.3.4.2, "Training Programs", and 5.2.4, "Process Safety - Training," SRD Volume II, Section 7 requires that the program address:

- continual training - SC 7.2-1, 7.2-3, 7.3-3
- qualification of personnel - SC 7.3-3
- records of training status - SC 7.2-4
- establishment of written procedures/instructions - SC 7.2-2, 7.2-5

Establishment of a Safety/Quality Program

The Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), Safety Criteria 1.0-1, requires the use of a comprehensive safety management program consistent with Top-Level Principle 5.1.1, “Process Safety Management”, and 5.1.2, “Process Safety Objective”. Safety Criterion 7.1-3 requires a safety framework be established to implement this Program consistent with Top-Level Principle 4.1.4.1, “Safety/Quality Culture”.

Establishment of a Quality Program is discussed above under the heading, “Quality Assurance Program”.

3.0 Determination of SSCs for the Implementation of Defense in Depth

The standards for prevention, control, and human aspects in Sections 2.2, 2.3, and 2.6 are primarily concerned with defense in depth sub-principles that minimize the potential of hazard initiation. In evaluating accidents that are postulated to occur despite implementation of preventive, control and human aspects, the sub-principles of mitigation and automatic systems must be considered.

The Implementing Standard for Safety Standards and Requirements Identification describes the process by which hazards and hazardous situations are identified and evaluated to determine hazard control strategies. Use of this Implementing Standard ensures that the defense in depth sub-principles are accounted for in the process of determining hazard control strategies. That process will identify SSCs that perform defense in depth as part of their safety function. The administrative controls that back up these SSCs will be developed prior to the introduction of hazardous materials into the facility.

Table 1 is the standard for implementing defense in depth by SSCs as part of the hazard control strategy; it defines the minimum number of SSCs and associated engineering requirements for the control of hazards of a particular severity.

Table 1 will be used in conjunction with the guidance in Section 2.0 to ensure that the preferred control solution addresses the strategies that protect the public and collocated workers; such SSCs will always be backed up by the human aspects of defense in depth discussed in Section 2.6.

The table lists the number and attributes of the physical barriers, as well as the application of the single failure criterion to SSCs that are required to adequately implement defense in depth for a given control strategy. Confirmation of the adequacy of implementation is achieved by meeting the numerical guidance stated in the third column. Consistent with the defense in depth sub-principles in Section 2.0, the control strategy should emphasize passive SSCs over active SSCs.

Hazard severities and target frequencies are the means to achieve adequate defense in depth in accordance with the tailored approach mandated by RL/REG 98-17, “Regulatory Unit Position on Tailoring for Safety.”

1st Column - SL (Severity Level)

Determination of hazard severity level is based on an assessment of unmitigated consequences. Severity levels are defined as SL-1 to SL-4, with SL-1 having the highest consequences.

2nd Column - Control Options for Implementation of Defense in Depth

A graded approach is reflected in the configuration requirements against each hazard severity level. The requirements are more stringent for defense in depth implementation for hazards of greater severity than for those of lesser severity.

Implementation of defense in depth requires that the single failure criterion be applied in a tailored fashion. For SL-1, application of the single failure criterion is mandatory. For SL-2, the single failure criterion shall be considered; that is, an objective assessment must be performed to determine the extent to which the single failure criterion will be incorporated into or be satisfied by design. The results and basis of this assessment shall be documented. Such documentation shall be retrievable and can be in the form of engineering studies, meeting minutes, reports, internal memoranda, etc. The single failure criterion is discussed in Section 2.1.

In addition to the single failure criteria in Table 1, diversity may also be implemented in the control strategy where hazards assessment reveals a common mode failure concern (see the Implementing Standard for Safety Standards and Requirements Identification, SRD Vol. II, Appendix A).

Implementation of defense in depth also requires that the provision of physical barriers be applied in a tailored fashion. In Table 1, provision of physical barriers refers to those that provide confinement against the release of hazardous materials, as opposed to barriers that protect against direct radiation. For SL-1 and SL-2, two or more independent physical barriers are required. For SL-3, at least one physical barrier shall be provided, and two or more independent physical barriers shall be considered; that is, an objective assessment must be performed to determine the extent to which physical barriers will be incorporated by the design. The results and basis of this assessment shall be documented. Such documentation shall be retrievable and can be in the form of engineering studies, meeting minutes, reports, internal memoranda, etc. For SL-4, physical design features and/or administrative controls per 10CFR 835.1001 shall be provided.

The graded approach is also reflected in the degree of confidence required commensurate with the hazard severity. The confidence is based on the standards and other attributes applicable to the particular control strategy. The Implementing Standard for Safety Standards and Requirements Identification describes selection of standards and other attributes applicable to control strategies.

3rd Column - Target Frequency (yr^{-1})

This column lists the target frequencies for each hazard severity level. The hazard severity level is a measure of the consequence from an unmitigated event - that is, an event in which both SSCs that prevent the accident and SSCs that mitigate the accident fail. After the preferred hazard control strategy has been identified, the event frequency - i.e., the product of the frequency of the initiating event and the probability that the control strategy will fail given the initiating event - will be conservatively estimated. (No credit is taken for administrative controls in calculating the initiating event frequency.) Verifying that the event frequency is less than the target frequency will provide confirmation that the chosen control strategy includes sufficient SSCs to adequately implement defense in depth in a graded approach.

The demonstration of having met the target frequencies may be based on either numerical analysis or engineering judgment. When appropriate, administrative controls alone may be credited as the controls that protect facility workers. The hazard assessment and control team shall assess the confidence in the frequency so determined, applying greater conservatism where engineering judgment is employed.

Table 1. Implementation of Defense in Depth by SSCs.

Severity Level	Control Options for Implementation of Defense in Depth	Target Frequency (yr^{-1})
SL-1	Two or more independent physical barriers. The single failure criterion shall be applied.	$< 10^{-6}$
SL-2	Two or more independent physical barriers. The single failure criterion shall be considered.	$< 10^{-4}$
SL-3	At least one physical barrier shall be provided. Two or more independent physical barriers shall be considered.	$< 10^{-2}$
SL-4	Physical design features and/or administrative controls per 10 CFR 835.1001	$< 10^{-1}$

4.0 Definitions

Definitions of the following terms were obtained from the referenced consensus standards. Minor wording differences among multiple references are ignored. In some cases, the definition of a term given in the referenced consensus standard has been tailored to the relative risks of the RPP-WTP and its anticipated associated hazards. Other wording differences in the definitions below from the cited consensus standards have been made to preserve consistency with terminology in other RPP-WTP safety documentation. Such differences are identified by presenting added words in Italics and by inserting double-brackets where words have been removed. Citation of a definition from a given consensus standard shall not be read to infer that other portions of the standard not specifically cited are being invoked.

Active component [SSC]. A component in which mechanical movement must occur to accomplish the [ ] safety function of the component (Ref. 5.5, 5.6)

Active failure. A malfunction, excluding passive failures, of a component that relies on mechanical movement to complete its intended [ ] safety function upon demand

Examples of active failures include the failure of a valve or check valve to move to its correct position, or the failure of a pump, fan, or diesel generator to start.

Spurious action of a powered component originating within its actuation or control system shall be regarded as an active failure unless the specific design features or operating restrictions preclude such spurious action. An example is the unintended energization of a powered valve to open or close (Ref. 5.5, 5.6, 5.8).

Administrative controls. Provisions relating to organization and management, procedures, record keeping, assessment, and reporting necessary to ensure safe operation of the facility.

Barrier. A control that has the function of maintaining confinement, that is preventing or mitigating the release of radioactive or hazardous material to the worker, public or the environment. This control can be an SSC that provides a physical barrier (e.g. vessel, shielding, and filtration) or an administrative control (e.g. training and procedures), which supplements the physical barriers.

Common cause failure. Dependent failures that are caused by a condition external to a system or set of components that make system or multiple component failures more probable than multiple independent failures (Ref. 5.4).

Common mode failure. Dependent failures caused by susceptibilities inherent in certain systems or components that make their failures more probable than multiple independent failures due to those components having the same design or design conditions that would result in the same level of degradation (Ref. 5.4).

Confinement. Physical barrier that prevents or mitigates the release of radioactive or hazardous material to the worker, public or the environment. The DOE nonreactor facility safety Implementation Guide identifies three kinds of confinement barriers - primary confinement, secondary confinement, and tertiary confinement (Ref. 5.3).

Control strategy. A set of generally-described provisions (barriers, dilution/dispersal, physical limitations on material quantities, administrative material controls, confinement, ventilation of flammable gas, etc.) and/or approaches (defense in depth, use of passive features, prevention, mitigation, etc.) which are intended to assure adequate control of a specific hazard and associated accidents in the context of the work (Ref. 5.4).

Defense in depth. The fundamental principle underlying the safety technology of the facility centered on several levels of protection including successive barriers preventing the release of radioactive materials to the workplace or the environment. Human aspects of defense in depth are considered to protect the integrity of the barriers, such as quality assurance, administrative controls, safety reviews, operating limits, personnel qualifications and training and safety program. Design provisions including both those for normal facility systems and those for systems important to safety help to: 1) prevent undue challenges to the integrity of the physical barriers; 2) prevent failure of a barrier if challenged; 3) where it exists, prevent consequential damage to multiple barriers in series; and 4) mitigate the consequences of accidents. Defense in depth helps to assure that two basic safety functions (controlling the process flow and confining the radioactive material) are preserved and that radioactive materials do not reach the worker, public or the environment (Ref. 5.4).

Design Basis Events. Postulated events providing bounding conditions for establishing the performance requirements of structures, systems and components that are necessary to: 1) ensure the integrity of the safety boundaries protecting the worker; 2) place and maintain the facility in a safe state indefinitely; or 3) prevent or mitigate the event consequences so that the radiological exposures to the general public or the workers would not exceed appropriate limits. The Design Basis Events also establish the performance requirements of the structures, systems, and components whose failure under Design Basis Event conditions could adversely affect any of the above functions (Ref. 5.4).

Detectable failures. [The following definition is considered to be specific to electrical, instrumentation and control systems.]

> Failures that can be identified through periodic testing or can be revealed by alarm or anomalous indication (Ref. 5.9).

Diversity. Use of different technologies, equipment, or design methods to perform a common function with the intent to minimize common cause failures (Ref. 5.13).

Engineered feature. A structure, system or component that contributes to the safe operation of the facility (Ref. 5.14).

Event. A condition that deviates from normal operation, i.e., an initiating occurrence plus single failure or coincident occurrence combination (Ref. 5.5, 5.6).

External Event. An event external to the RPP-WTP caused by (1) a natural hazard (e.g., earthquake, flood, lightning, or range fire) or (2) a human-induced event (e.g., transportation or nearby industrial activity).

Human factors engineering (HFE). An interdisciplinary science and technology concerned with the process of designing for human use (Ref. 5.12).

Important to Safety. Structures, systems and components that serve to provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the workers and the public. It encompasses the broad class of facility features addressed (not necessarily explicitly) in the top-level radiological nuclear, and process safety standards and principles that contribute to the safe operation and protection of workers and the public during all phases and aspects of facility operations (i.e., normal operation as well as accident mitigation).

This definition includes not only those structures, systems and components that perform safety functions and traditionally have been classified as safety class, safety-related or safety grade, but also those that place frequent demands on or adversely affect the performance of safety functions if they fail or malfunction, i.e., support systems, subsystems and components. Thus, these latter structures, systems, and components would be subject to applicable top-level radiological, nuclear and process safety standards and principles to a degree commensurate with their contribution to risk. In applying this definition, it is recognized that during the early stages of the design effort all significant systems interactions may not be identified and only the traditional interpretation of important to safety, i.e., safety-related may be practical. However, as the design matures and results from risk assessments identify vulnerabilities resulting from non-safety-related equipment, additional structures, systems and components should be considered for inclusion within this definition (Ref. 5.4).

Independence. The state in which there is no mechanism by which any single design basis event, such as a flood, can cause redundant equipment to be inoperable (Ref. 5.10).

Initiating occurrence/event. A single occurrence and its consequential effects that place the plant or some portion of the plant in an off-normal condition. An initiating occurrence/event is not the single failure defined elsewhere herein. An initiating occurrence can be *an internal event or an external event* (Ref. 5.5, 5.6, 5.8).

The first event in an event sequence. Can result in an accident unless engineered protection systems or human actions intervene to prevent or mitigate the accident (Ref. 5.15).

Internal Event. An occurrence related to structure, system, and component performance or human action, or an occurrence external to the system but within the RPP-WTP that causes upset of a structure, system, or component.

Limiting design requirements. The limiting value of a design parameter that ensures that the consequences of any event do not result in:

- Violation of plant nuclear safety criteria, including off-site radiological dose criteria, or
- Unacceptable degradation of plant components that are required to mitigate the consequences of an event.

(A single event may have more than one limiting design requirement.) (Ref. 5.7)

Long term. *For fluid systems*, the long term is defined as that period of *important to safety* fluid system operation following the short term during which the safety function of the system is required (Ref. 5.8).

Passive component. A component that is not an active component (Ref. 5.5, 5.6).

Passive failure. The blockage of a process flow path or failure of a component to maintain its structural integrity or stability, such that it cannot provide its intended [ ] safety function upon demand (Ref. 5.5, 5.6, 5.8).

Primary confinement. Provides confinement of hazardous material to the vicinity of its processing. This confinement is typically provided by piping, tanks, glove boxes, encapsulating material, and the like, along with any offgas systems that control effluent from the primary confinement (Ref. 5.3).

Redundant equipment or system. A system or component that duplicates the essential functions of another system or component to the extent that either may perform the required function, regardless of the state of operation or failure of the other (Ref. 5.9, 5.10).

Safety function. Any function that is necessary to ensure: 1) the integrity of the boundaries retaining the radioactive materials; 2) the capability to place and maintain the facility in a safe state; or 3) the capability to prevent or mitigate the consequences of facility conditions that could result in radiological exposures to the general public or workers in excess of appropriate limits (Ref. 5.4).

Secondary confinement. Consists of a cell or enclosure surrounding the process material or equipment along with any associated ventilation exhaust systems from the enclosed area. Except in the case of housing glove-box operations, the area inside this barrier is usually unoccupied (e.g., canyons, hot cells); it provides protection for operating personnel (Ref. 5.3).

Shall, should and may. The word “shall” is used to denote a requirement; the word “should” is used to denote a recommendation; and the word “may” is used to denote permission, neither a requirement nor a recommendation (Ref. 5.5, 5.6, 5.8).

The word “shall” denotes actions that must be performed… The word “should” is used to indicate recommended practice (Ref. 5.3, based on DOE-STD-1075-94).

Shall [be] consider[ed]. An objective assessment must be performed to determine the extent to which the single failure criterion will be incorporated into or be satisfied by design. The results and basis of this assessment shall be documented. Such documentation shall be retrievable and can be in the form of engineering studies, meeting minutes, reports, internal memoranda, etc. (Ref. 5.16).

Short term. *For fluid systems*, the short term is defined as that period of operation up to 24 hours following an initiating event [ ] (Ref. 5.8).

Single failure. A random failure and its consequential effects, in addition to an initiating occurrence, that result in the loss of capability of a component to perform its intended [ ] safety function(s) (Ref. 5.5, 5.6).

Single failure criterion. [Two definitions are provided below. The following definition applies to fluid (i.e., liquid and gas) systems.]

> Fluid [ ] systems are considered to be designed against an assumed single failure if neither (1) a single failure of any active component (assuming passive components function properly), nor (2) a single failure of any passive component (assuming active components function properly) results in a loss of the capability of the system to perform its [ ] safety function (Ref. 5.5, 5.6).

[The following statement of the "single failure criterion" applies to electrical and instrumentation and control systems.]

The *important to* safety systems shall perform all required safety functions for a design basis event in the presence of the following:

1. Any single detectable failure within the *important to* safety systems concurrent with all identifiable but non-detectable failures
2. All failures caused by the single failure
3. All failures and spurious system actions that cause, or are caused by, the design basis event requiring the safety function

The single failure could occur prior to, or at any time during, the design basis event for which the *important to* safety system is required to function (Ref. 5.9).

5.0 References

5.1 10 CFR 830.120, Quality Assurance Requirements

5.2 DOE O 420.1, Facility Safety, Chg 2, 10/24/96

5.3 Implementation Guide for Nonreactor Nuclear Safety Criteria and Explosives Safety Criteria, Rev. G (Draft), September 1995

5.4 DOE/RL-96-0006, Revision 1, Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for TWRS Privatization Contractors

5.5 ANSI/ANS-51.1-1983, Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants. American Nuclear Society

5.6 ANSI/ANS-52.1-1983, Nuclear Safety Criteria for the Design of Stationary Boiling Water Reactor Plants. American Nuclear Society

5.7 ANSI/ANS-58.8-1994, Time Response Design Criteria for Safety-Related Operator Actions. American Nuclear Society

5.8 ANSI/ANS-58.9-1981, Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems. American Nuclear Society

5.9 IEEE Std 379-1994, IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems. IEEE Power Engineering Society

5.10 IEEE Std 384-1992, IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits. IEEE Power Engineering Society

5.11 IEEE Std 603-1991, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations. IEEE Power Engineering Society

5.12 IEEE Std 1023-1988, IEEE Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations.

5.13 ISA-S84.01-1996, Application of Safety Instrumented Systems for the Process Industries. Instrument Society of America.

5.14 Integrated Safety Management Plan. 24590-WTP-ISMP-ESH-01-001. Bechtel National, Inc., Richland, WA

5.15 Guidelines for Hazards Evaluation Procedures, Second Edition with Worked Examples. Center for Process Chemical Safety, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York, 1992.

5.16 DOE O 6430.1A, General Design Criteria, 6 April 1989

6.0 Tailoring of Consensus Standards Used in the Implementing Standard for Defense in Depth

The following subsections summarize the RPP-WTP contractor's tailoring of the consensus standards invoked by this Implementing Standard for Defense in Depth.

6.1 DOE O 420.1, Facility Safety (Ref. 5.2)

Terminology

- Section 4.1.1.2, 1st paragraph, last sentence: Phrase "…workers, including those at adjacent facilities…" is interpreted for RPP-WTP to mean "…workers and collocated workers…"

Applicability

- The only portion of DOE O 420.1 that is being invoked by this Implementing Standard for Defense in Depth is Section 4.1.1.2, the first three paragraphs.

6.2 Implementation Guide for Nonreactor Nuclear Safety Criteria and Explosives Safety Criteria (Ref. 5.3)

Terminology

- By virtue of cross-references within the DOE Implementation Guide (IG), reference is made to "safety class" and "safety significant" SSCs. The RPP-WTP project uses the term "important to safety", which encompasses both "safety class" and "safety significant".
- "Critical safety function" in the DOE IG is interpreted to more broadly read "…significant public, worker and collocated worker impact".

Applicability

- The only portion of the DOE "420.X" Implementation Guide that is being invoked by this Implementing Standard for Defense in Depth is Section 2.3, except the last paragraph.
- Section 2.3 of the DOE IG contains internal cross-references to subsections 5.2.1, 5.2.2.1 and 5.2.2.2, which list typical codes for structures, ventilation systems, and process equipment that provide a confinement function. Section 2.4.2 of this Implementing Standard lists the SRD Safety Criteria that will be applied to SSCs comprising confinement.
- Section 2.3 of the DOE IG contains an internal cross-reference to subsection 5.2.1, which further cites section 4.4 of DOE O 420.1 and section 3.3 of the DOE IG for criteria for natural phenomena hazards (NPH). For the RPP-WTP, NPH criteria are provided in SRD Safety Criteria SC 4.1-3 and SC 4.1-4.

6.3 ANSI/ANS-58.8-1994, Time Response Design Criteria for Safety-Related Operator Actions (Ref. 5.7)

Terminology

- "Safety-related" is interpreted for RPP-WTP to mean "important to safety" or "ITS".
- "Safety-related function" is interpreted for RPP-WTP to mean "safety function" as defined in DOE/RL-96-0006, Rev. 1.

Non-Applicability

- Assumption (1) of section 1.3 does not apply. Single failure criteria for the RPP-WTP project are given in the consensus standards invoked and tailored by this Implementing Standard (ANSI/ANS-58.9-1981 and IEEE 379-1994).
- Assumption (4) of section 1.3 does not apply. The operators will be qualified in accordance with the RPP-WTP training program, per Safety Requirements Document Volume II (24590-WTP-SRD-ESH-01-001-02), Section 7.2.
- "Automatic reactor trip…" does not apply.

6.4 ANSI/ANS-58.9-1981, Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems (Ref. 5.8)

Terminology

- "Containment" or "containment vessel" is interpreted to mean "confinement".
- "Seismic Category I standards" is interpreted as seismic requirements for a SSC with a seismic safety function per SRD Volume II (24590-WTP-SRD-ESH-01-001-02) Safety Criterion 4.1-3 for the RPP-WTP.
- "Safety related" is interpreted for RPP-WTP to mean "important to safety" or "ITS". Conversely, "non-safety-related" means "non-ITS".
- "Technical specification(s)" is interpreted to mean "Technical Safety Requirements" or "TSR(s)".
- "Condition I" is interpreted for RPP-WTP to mean "normal operation".
- "Safety-related function" is interpreted for RPP-WTP to mean "safety function" as defined in DOE/RL-96-0006, Rev. 1.
- In definition of "single failure", reference [1] does not apply to RPP-WTP.
- Safety classes 1, 2, and 3 (section 4.5) are interpreted to be important to safety systems.

Non-Applicability

- For RPP-WTP, the need for emergency onsite power will be ascertained in accordance with the DOE/RL-96-0004 process as part of determining hazard control strategies.
- In the definition of "short term" (section 2), everything after "…up to 24 hours following an initiating event" applies to nuclear power reactor plants and is therefore not applicable to RPP-WTP.
- Sections 3.1 through 3.3 of ANSI/ANS 58.9 are not applicable to the RPP-WTP. Applicability of the single failure criteria to the work and hazards presented by the RPP-WTP is described in Section 3.0 of this Implementing Standard.
- Reactor-specific regulations (e.g., 10 CFR 50 Appendix A) are not applicable to RPP-WTP (see Section 1, 1st paragraph).
- References to a reactor "unit", "safe shutdown", and "loss of coolant accident" are nuclear reactor plant-specific and, therefore, do not apply to RPP-WTP.
- Sections 3.1 through 3.3 are reactor-specific and do not apply to RPP-WTP.

6.5 IEEE STD 379-1994, IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems (Ref. 5.9)

Terminology

- BNI uses the definitions of the following terms given in DOE/RL-96-0006, rather than those in section 3 of the consensus standard:
  - Common-cause failure
  - Design basis events
  - Safety function
- "Safety system" is interpreted for RPP-WTP to mean "important to safety system". Consequently, "important to safety system" is interpreted to mean a system that performs a safety function, as defined in DOE/RL-96-0006.
- "Containment" or "containment vessel" is interpreted to mean "confinement" for the RPP-WTP.

Applicability

- Applicability of the single failure criteria to the work and hazards presented by the RPP-WTP is described in Section 3.0 of this Implementing Standard.
- Nuclear reactor plant-specific terms such as reactor "unit", "reactor trip system" power, control rods, "safety injection", "core spray", and "low pressure coolant injection" do not apply to RPP-WTP.

6.6 IEEE Std 603-1991, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations (Ref. 5.11)

Terminology

- The definition of "administrative controls" in section 2 of the consensus standard is understood as being consistent with the definition given in section 4.0 of this Implementing Standard.
- The definition of "Class 1E" is interpreted for the RPP-WTP as follows: "The classification of the electric equipment and systems that perform a safety function." The note following the definition of "Class 1E" in the consensus standard is retained for RPP-WTP.
- BNI uses the definitions of the following terms given in DOE/RL-96-0006, rather than those in section 3 of the consensus standard:
  - Design basis events
  - Safety function
- "Safety system" is interpreted for RPP-WTP to mean "important to safety system". Consequently, "important to safety system" is interpreted to mean a system that performs a safety function, as defined in DOE/RL-96-0006.
- "Containment" or "containment vessel" is interpreted to mean "confinement" for the RPP-WTP.
- "Nuclear power generating stations" is interpreted to mean a nuclear facility such as RPP-WTP.

Non-Applicability

- Nuclear reactor plant-specific terms such as reactor "unit", "emergency reactor shutdown", "reactor heat removal", do not apply to RPP-WTP.

6.7 IEEE STD 1023-1988, IEEE Guide for the Application of Human Factors Engineering to Systems, Equipment, and Facilities of Nuclear Power Generating Stations (Ref. 5.12)

Terminology

- “Nuclear power generating stations” is interpreted to mean a nuclear facility such as RPP-WTP.

Non-Applicability

- Application of the formal human factors engineering process described in subsection 6.1.1 of IEEE Std 1023-1988 is tailored to the work and hazards presented by the RPP-WTP in subsection 2.6.2 of this Implementing Standard.

- Section 6.1.1.12 and 6.1.1.18 of IEEE Standard 1023-1998 recommends the use of a separate plant simulator or physical mockup for human factors engineering. As discussed in subsection 2.6.2 of this Implementing Standard, the RPP-WTP contractor does not currently plan to construct a separate plant simulator or physical mockup and these sections are, therefore, not applicable to RPP-WTP.

6.8 ISA-S84.01-1996, Safety Instrumented Systems for the Process Industries (Ref. 5.13)

Terminology

- The definition of “common-cause failure” given in DOE/RL-96-0006 is used, rather than that in section 3 of the consensus standard.
- “Safety Instrumented System (SIS)” is interpreted to refer to any instrumentation and control system in the RPP-WTP that is important to safety, as defined in DOE/RL-96-0006.

Appendix C

Implementing Standards

CONTENTS

1.0 ISO 10007, Quality Management - Guidelines for Configuration Management

Revision: First Edition, April 15, 1995
Sponsoring Organization: International Organization for Standardization

RPP-WTP Specific Tailoring

The following tailoring of ISO 10007 is required for use by the RPP-WTP contractor as an Implementing Standard for Configuration Management.

Page 1, Section 1 Scope

Second paragraph, last sentence. Replace this sentence with:

"For RPP-WTP ISO 10007 amplifies on the configuration management elements found in the Quality Assurance Program, BNFL-5193-QAP-01."

Justification: For RPP-WTP the approved QAP defines the quality management and quality system elements.

Page 1, Section 2 Normative References

Delete reference to the ISO 10011 series of standards.

Justification: As discussed for Section 8, for RPP-WTP the approved QAP defines the principles, criteria, and practices for the configuration management system audit.

Page 7, Section 7.7 Configuration Management Plant (CMP)

Delete second paragraph

Justification: This paragraph addresses activities outside the RPP-WTP project workscope or control (i.e., multiple projects, multi-level contracts, and customer configuration management plans).

Page 8, Section 8 Configuration Management System Audit

Revise the last paragraph to read:

"Principles, criteria, and practices of the CM system audit should comply to the RPP-WTP QAPIP."

Justification: For RPP-WTP the approved QAP defines the principles, criteria, and practices for the conduct of audits and self-assessments.

Page 11 and 12, Annex B

Delete.

Justification: Although provided only as information, as noted in Section 1 above, the ISO 9000 Series of Standards are not being implemented at RPP-WTP and this Annex is therefore removed to reduce potential confusion to non-applicable cross references.

2.0 DOE-STD-1020-94, "Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities"

Revision: Change Notice #1 dated 1/96 and DOE Newsletter dated 1/22/98 (Interim Advisory on Straight Winds and Tornados)

Sponsoring Organization: DOE

RPP-WTP Specific Tailoring

The following tailoring of DOE-STD-1020-94 is required for use by the RPP-WTP contractor as an Implementing Standard for seismic analysis and design.

Page 1-6, Section 1.3 Evaluation of Existing Facilities

Delete this section.

Justification: This section deals with existing facilities and the RPP-WTP is a new facility.

Page 2-1, Section 2.2 General Approach for Seismic Design and Evaluation

Use 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Design PC-3 (Seismic Category I) SSCs for the elastic seismic response to DBE per Section 3.7.2 of NRC NUREG-0800, Rev. 3 (Draft) with no credit for inelastic energy absorption. Note: Credit for inelastic energy absorption is allowed in the design of PC-3 (Seismic Category II) SSCs.

Justification: This change is made for consistency with NRC acceptance criteria.

Use ASCE 4-98 (Draft) in lieu of ASCE 4-86.

Justification: ASCE 4-98 (Draft) is more current.

Page 2-6, Section 2.3 Seismic Design and Evaluation of Structures, Systems, and Components

Perform performance categorization of SSCs per SRD Safety Criteria 4.1-3 and 4.1-4 in lieu of DOE-STD-1021-93.

Justification: DOE-STD-1021-93 is inconsistent with the top-level safety principles in DOE/RL-96-0006. The functions of this standard are implemented by SRD Safety Criteria 4.1-3 and 4.1-4 and Appendix A to Volume II of the SRD.

Page 2-8, Section 2.3.1 Performance Category 1 and 2 Structures, Systems, and Components

Use 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Page 2-12, Section 2.3.2 Performance Category 3 and 4 Structures, Systems, and Components

Disregard the requirements for PC-4 SSCs.

Justification: There are no PC-4 SSCs at the RPP-WTP.

Design PC-3 (Seismic Category I) SSCs for the elastic seismic response to DBE per Section 3.7.2 of NRC NUREG-0800, Rev. 3 (Draft) with no credit for inelastic energy absorption. Note: Credit for inelastic energy absorption is allowed in the design of PC-3 (Seismic Category II) SSCs.

Justification: This change is made for consistency with NRC acceptance criteria.

Use ACI 349 for design of reinforced concrete in lieu of UBC.

Justification: This change is made for consistency with NRC acceptance criteria contained in Section 3.8.4 of NUREG-0800, Rev. 2 (Draft).

Use ANSI/AISC N690 for design of structural steel in lieu of UBC.

Justification: This change is made for consistency with NRC acceptance criteria contained in Section 3.8.4 of NUREG-0800, Rev. 2 (Draft).

Page 2-15, Section 2.3.3 Damping Values for Performance Category 3 and 4 Structures, Systems, and Components

Use ASME Code Case N-411 damping value for piping in lieu of those shown in Table 2-3.

Justification: This value is acceptable to the NRC for nuclear power plants.

Page 2-18, Section 2.4.1 Equipment and Distribution Systems

Perform seismic design of PC-1 and -2 elements of structures and equipment per the provisions of 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Page 2-22, Section 2.4.2 Evaluation of Existing Facilities

Delete this section.

Justification: This section deals with existing facilities and the RPP-WTP is a new facility.

Page 2-24, Section 2.5 Summary of Seismic Provisions

Disregard the requirements for PC-4 SSCs.

Justification: There are no PC-4 SSCs at the RPP-WTP.

Design PC-3 (Seismic Category I) SSCs for the elastic seismic response to DBE per Section 3.7.2 of NRC NUREG-0800, Rev. 3 (Draft) with no credit for inelastic energy absorption. Note: Credit for inelastic energy absorption is allowed in the design of PC-3 (Seismic Category II) SSCs.

Justification: This change is made for consistency with NRC acceptance criteria.

Use the seismic provisions in Table 2-5 concerning PC-3 SSCs except that the structural capacity is to be based on code ultimate strength or allowable behavior level.

Justification: Limit-state level method of determining the structural capacity is more appropriate for evaluation of existing facilities (the RPP-WTP is a new facility).

Page 3-1, Section 3.1 Introduction

Perform performance categorization of SSCs per SRD Safety Criteria 4.1-3 and 4.1-4 in lieu of DOE-STD-1021-93.

Justification: DOE-STD-1021-93 is inconsistent with the top-level safety principles in DOE/RL-96-0006. The functions of this standard are implemented by SRD Safety Criteria 4.1-3 and 4.1-4 and Appendix A to Volume II of the SRD.

Page 3-2, Section 3.2 Wind Design Criteria

Use peak gust speed values contained in Attachment "A" of DOE Interim Advisory dated 1/22/98 in lieu of fastest-mile wind speeds shown in Table 3-2; also, per DOE Interim Advisory, use an importance factor for PC-2 SSCs of 1.0 in lieu of 1.07 indicated in Table 3-1.

Justification: The Newsletter was issued by DOE as an interim measure for use with DOE-STD-1020-94 until such time as the standard is revised.

Page 3-5, Section 3.2.1 Performance Category 1

Design structural steel PC-1 structures per AISC Manual of Steel Construction, Allowable Stress Design, Ninth edition.

Justification: The AISC code is preferred to the UBC because it is a national consensus code.

Design reinforced concrete PC-1 structures per ACI 318-99.

Justification: The ACI 318 code is preferred to the UBC because it is a national consensus code.

Page 3-6, Section 3.2.2 Performance Category 2

Design structural steel PC-2 structures per AISC Manual of Steel Construction, Allowable Stress Design, Ninth edition.

Justification: The AISC code is preferred to the UBC because it is a national consensus code.

Design reinforced concrete PC-2 structures per ACI 318-99.

Justification: The ACI 318 code is preferred to the UBC because it is a national consensus code.

Page 3-6, Section 3.2.3 Performance Category 3

Design structural steel PC-3 structures per ANSI/AISC N690-94.

Justification: This change is made for consistency with NRC acceptance criteria contained in Section 3.8.4 of NUREG-0800, Rev. 2 (Draft).

Design reinforced concrete PC-3 structures per ACI 349-97.

Justification: This change is made for consistency with NRC acceptance criteria contained in Section 3.8.4 of NUREG-0800, Rev. 2 (Draft).

Disregard requirements for tornado design.

Justification: Tornado is not a credible NPH at the RPP-WTP site.

Page 3-11, Section 3.2.4 Performance Category 4

Delete this section.

Justification: There are no PC-4 SSCs at the RPP-WTP.

Page 3-13, Section 3.3 Evaluation of Existing SSCs

Delete this section.

Justification: This section deals with existing facilities and the RPP-WTP is a new facility.

Page 4-1, Section 4.0 Flood Design and Evaluation Criteria

Disregard criteria for the design of SSCs for river flooding.

Justification: River flooding is not a credible NPH at the RPP-WTP site, and only the criteria dealing with local precipitation that affects roof design and site drainage are applicable to the RPP-WTP design.

Page 4-4, Section 4.1.2 Flood Evaluation Process

Perform performance categorization of SSCs per SRD Safety Criteria 4.1-3 and 4.1-4 in lieu of DOE-STD-1021-93.

Justification: DOE-STD-1021-93 is inconsistent with the top-level safety principles in DOE/RL-96-0006. The functions of this standard are implemented by SRD Safety Criteria 4.1-3 and 4.1-4 and Appendix A to Volume II of the SRD.

Page 4-12, Section 4.2.4 Performance Category 4

Delete this section.

Justification: There are no PC-4 SSCs at the RPP-WTP.

Page 4-13, Section 4.3.3 Site Drainage and Roof Design

Use 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Page 4-15, Section 4.4 Considerations for Existing Construction

Delete this section.

Justification: This section deals with existing facilities and the RPP-WTP is a new facility.

Page 4-16, Section 4.5 Probabilistic Flood Risk Assessment

Do not perform a probabilistic flood risk assessment of the RPP-WTP site.

Justification: UCRL-21069, “Probabilistic Flood Hazard Assessment for the N Reactor, Hanford, Washington”, July 1988, contains a probabilistic flood risk assessment of the N reactor site. The RPP-WTP site is close to the N Reactor site (about 10 miles away) and further away from the Columbia River. Therefore, the N Reactor flood assessment may be used and no assessment of the RPP-WTP site is required.

Page B-4, App. B, Section B.2 Graded Approach, Performance Goals, and Performance Categories

Perform performance categorization of SSCs per SRD Safety Criteria 4.1-3 and 4.1-4 in lieu of DOE-STD-1021-93.

Justification: DOE-STD-1021-93 is inconsistent with the top-level safety principles in DOE/RL-96-0006. The functions of this standard are implemented by SRD Safety Criteria 4.1-3 and 4.1-4 and Appendix A to Volume II of the SRD.

Page B-8, App. B, Section B.3 Evaluation of Existing Facilities

Delete this section.

Justification: This section deals with existing facilities and the RPP-WTP is a new facility.

Page C-1, App. C, Section C.1 Introduction

Perform performance categorization of SSCs per SRD Safety Criteria 4.1-3 and 4.1-4 in lieu of DOE-STD-1021-93.

Justification: DOE-STD-1021-93 is inconsistent with the top-level safety principles in DOE/RL-96-0006. The functions of this standard are implemented by SRD Safety Criteria 4.1-3 and 4.1-4 and Appendix A to Volume II of the SRD.

Page C-19, App. C, Section C.3.2 Earthquake Ground Motion Response Spectra

Disregard Section C.3.2.1 discussion and Table C-4. Follow 1997 UBC for the RPP-WTP design.

Justification: Section C.3.2.1 discussion and Table C-4 are based on 1994 UBC; the 1997 UBC is more current.

Page C-27, App. C, Section C.4 Evaluation of Seismic Demand (Response)

Use 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Page C-29, App. C, Section C.4.1 Dynamic Seismic Analysis

Use ASCE 4-98 (Draft) in lieu of ASCE 4-86.

Justification: ASCE 4-98 (Draft) is more current.

Page C-31, App. C, Section C.4.2 Static Force Method of Seismic Analysis

Use 1997 UBC in lieu of 1994 UBC.

Justification: 1997 UBC is more current.

Page C-32, App. C, Section C.4.3 Soil-Structure Interaction

Use ASCE 4-98 (Draft) in lieu of ASCE 4-86.

Justification: ASCE 4-98 (Draft) is more current.

Page C-38, App. C, Section C.4.4 Analytical Treatment of Energy Dissipation and Absorption

Design PC-3 (Seismic Category I) SSCs for the elastic seismic response to DBE per Section 3.7.2 of NRC NUREG-0800, Rev. 3 (Draft) with no credit for inelastic energy absorption. Note: Credit for inelastic energy absorption is allowed in the design of PC-3 (Seismic Category II) SSCs.

Justification: This change is made for consistency with NRC acceptance criteria.

Page C-52, App. C, Section C.5.1 Capacity Approach

Use ACI 349 for design of reinforced concrete in lieu of UBC.

Justification: This change is made for consistency with NRC acceptance criteria contained in Section 3.8.4 of NUREG-0800, Rev. 2 (Draft).

Use ANSI/AISC N690 for design of structural steel in lieu of UBC.

Justification: This change is made for consistency with NRC acceptance criteria contained in Section 3.8.4 of NUREG-0800, Rev. 2 (Draft).

Page C-62, App. C, Section C.7 Special Considerations for Existing Facilities

Delete this section.

Justification: This section deals with existing facilities and the RPP-WTP is a new facility.

Page C-66, App. C, Section C.9 Alternate Seismic Mitigation Measures

Delete this section.

Justification: Seismic base isolation is not planned to be used in the RPP-WTP design.

Page D-3, App. D, Section D.3 Load Combinations

Design structural steel PC-1 and PC-2 structures per AISC Manual of Steel Construction, Allowable Stress Design, Ninth edition.

Justification: The AISC code is preferred because it is a national consensus code.

Design reinforced concrete PC-1 and PC-2 structures per ACI 318-99.

Justification: The ACI 318 code is preferred because it is a national consensus code.

Design structural steel PC-3 SSCs structures per ANSI/AISC N690-94.

Justification: This change is made for consistency with NRC acceptance criteria contained in Section 3.8.4 of NUREG-0800, Rev. 2 (Draft).

Design reinforced concrete PC-3 SSCs structures per ACI 349-97

Justification: This change is made for consistency with NRC acceptance criteria contained in Section 3.8.4 of NUREG-0800, Rev. 2 (Draft).

3.0 ANSI/AISC N690, "Specification for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities"

Revision: 1994
Sponsoring Organization: American National Standards Institute/American Institute of Steel Construction

RPP-WTP Specific Tailoring

The following tailoring of ANSI/AISC N690 is required for use by the RPP-WTP contractor as an Implementing Standard for structural design.

Page 22, Section Q1.5.7.1 Primary Stresses

Revise the stress limit coefficients for compression in Table Q1.5.7.1 as follows:

- 1.3 instead of 1.5 [stated in footnote (c)] in load combinations 2, 5, and 6
- 1.4 instead of 1.6 in load combinations 7, 8, and 9
- 1.6 instead of 1.7 in load combination 11

Justification: These changes are made for consistency with the NRC requirements of Appendix F of Section 3.8.4 of NUREG-0800 (Draft Rev. 2).

Page 22, Section Q1.5.7.1 Primary Stresses

Delete the following load combinations:

4. D + L + Eo
6. D + L + Ro + To + Eo

Justification: These load combinations are requried for evaluation of an Operation Basis Earthquake (OBE). The WTP project has not identified an OBE event.

4.0 DOE G-420.1/G-440.1, Implementation Guide for Use with DOE Orders 420.1 and 440.1, Fire Safety Program*

Revision: September 30, 1995

Sponsoring Organization: U. S. Department of Energy

RPP-WTP Specific Tailoring

The following tailoring of DOE G-420.1/G-440.1 is required for use by the RPP-WTP Project as an implementing standard for fire safety.

Section III.5.0

Add the following words at the end of the paragraph: "The applicable building code for the RPP-WTP Project is the 1997 Uniform Building Code (UBC)."

Justification: To clarify that the code in effect at the time that facility design commenced was the 1997 UBC.

Section III.6.3

Revise to read "Automatic fire extinguishing systems in all areas subject to loss of safety class systems, significant life safety hazards, or unacceptable program interruption. The FHA may justify the omission of such systems based on safety considerations as approved by the AHJ.

Justification: The addition is consistent with governing Safety Criterion 4.5-4, which requires automatic fire suppression "unless the Fire Hazards Analysis dictates otherwise". It is also consistent with the DOE equivalency concept described in DOE G-420.1/G-440.1 Section II.

Section IV.4.5

Change "Description of critical process equipment" to "Identification of Important-To-Safety Equipment".

Justification: The term "critical process equipment" is not well defined for the RPP-WTP Project. By contrast the term "Important-to-Safety" is defined by the DOE regulatory documents, such as DOE/RL-96-0004. Identification of Important-to-Safety equipment is more meaningful and is consistent with the CAR Guidance (RL/REG-99-05).

5.0 DOE-STD-1066-97, Fire Protection Design Criteria

Revision: March 1997

Sponsoring Organization: U. S. Department of Energy

RPP-WTP Specific Tailoring

The following tailoring of DOE-STD-1066-97 is required for use by the RPP-WTP Project as an implementing standard for fire safety.

Section 9.5.1

Add the following words: "The fire resistance of special or unique penetration assemblies, such as lead glass windows and shield wall penetrations, may be based on past qualification testing or an equivalency evaluation."

Justification: The RPP-WTP facility is expected to have unique penetration configurations that may be impractical to test. This change clarifies that alternate approaches that provide a comparable level of safety, as described in Section 1 of DOE-Std-1066-97, may be used.

Section 10.4

Add the following words: "The 75-foot travel distance may be exceeded in areas not normally occupied by personnel, where plant equipment alone is located".

Justification: If an area is not normally occupied an accidental breach of a primary confinement system cannot expose personnel to radioactive material.

Section 10.6.3:

Delete the statement that: "In addition, for explosives environments, exits should reflect the criteria contained in the DOE Explosives Safety Manual (DOE M 440.1-1)."

Justification: The DOE Explosives Safety Manual applies to environments involving munitions, and is not applicable to the RPP-WTP.

Section 11.4

Revise this section to read: “Where required by the SAR, critical facilities should be served by dedicated, redundant electric power services. External to the buildings served, the two services should be separated by 4-hour fire-rated construction and should be served from separate sources.

Justification: The term “power services” is more consistent with the sentence that follows. The changes clarify that the requirement applies to site power supplies, not to cable routing within the buildings served.

Section 12.4

Delete this section.

Justification: This section is not applicable because there will be no gas-fired process furnaces in the RPP-WTP.

Section 13

Delete all except Subsection 13.1

Justification: The RPP-WTP is not a Plutonium Processing and Handling Facility, a Plutonium Storage Facility, an Enriched Uranium Storage Facility, a Uranium Processing and Handling Facility, a Fuel Reprocessing Facility, or a Uranium Conversion and Recovery Facility.

6.0 NFPA 801, Standard for Facilities Handling Radioactive Materials

Revision: 1995 edition

Sponsoring Organization: National Fire Protection Association

RPP-WTP Specific Tailoring

The following tailoring of NFPA 801-95 is required for use by the RPP-WTP project as an implementing standard for fire safety.

Section 3-8

Replace entire section with the text of the same section from the 1998 version of NFPA 801.

Justification: The NFPA standard was revised in recognition of the impracticality of using only noncombustible surface finishes in areas processing or storing radioactive materials. Conformance with the revised standard will permit the use of limited combustible interior finishes.

Section 6.1.1

Change the code edition for NFPA 70 from 1993 to 1996 and the code edition for NFPA 780 from 1992 to 1995.

Justification: SRD safety criteria 4.3-2 and 4.4-12 reference these more recent editions of NFPA 70 and NFPA 780 as implementing standards. This change resolves the conflict with NFPA 801.

7.0 ACI 349, Code Requirements for Nuclear Safety-Related Concrete Structures

Revision: 2001
Sponsoring Organization: American Concrete Institute

WTP Specific Tailoring

The following tailoring of ACI 349-01 is required for use by the WTP contractor as an implementing standard for structural design.

Chapter 21

Replace Chapter 21 of ACI 349-01 with Chapter 21 of ACI 318-99.

Justification: Chapter 21 of ACI 349-01 is based on criteria from ACI 318-95. The American Concrete Institute completed a major revision of ACI 318 between the years 1995 and 1999 with respect to seismic proportioning and detailing. The WTP project wishes to adopt the most current methodology as presented in ACI 318-99 in lieu of that presented in ACI 349-01 Chapter 21.

8.0 ACI 318, Building Code Requirements for Structural Concrete and Commentary

Revision: 1999
Sponsoring Organization: American Concrete Institute

WTP Specific Tailoring

The following tailoring of ACI 318-99 is required for use by the WTP contractor as an implementing standard for design of reinforced concrete for Seismic Category III SSCs, as noted.

Chapter 9, Section 9.2 Required Strength

The following additional load combinations from the *Uniform Building Code*, 1997, section 1612.2.1, shall be included in the load combinations evaluated for design of reinforced concrete:

Equation (12-5): $1.2D + 1.0E + (f_1L + f_2S)$

Equation (12-6): $0.9D \pm (1.0E \text{ or } 1.3W)$

Justification: The additional load combinations implemented are not identified in the ACI load combinations. These combinations are evaluated to ensure adequate equivalency with commercial design in accordance with the UBC.

Chapter 21, Section 21.2.1.3

Seismic detailing requirements for "moderate seismic risk" will be used.

Justification: The "moderate seismic risk" classification is consistent with the Seismic Category III, which is an important facility in seismic zone 2B.

General (no specific chapter)

Design of concrete anchorage will follow the requirements of PCA Publication EB 080.01, *Strength Design of Anchorage to Concrete*.

Justification: This design standard represents the current industry approach to design of concrete embedments. This design method has been adoped by ACI 349, committee and used in the 2001 edition for Appendix B. The load factors are lower than those identified for safety related structures applicable to higher seismic classification. The load factors in this publication are appropriate for use in important commercial structures commensurate with SC-III.

9.0 AISC M016, Manual of Steel Construction, Allowable Stress Design (ASD)

Revision: 9th Edition
Sponsoring Organization: American Institute of Steel Construction

WTP Specific Tailoring

The following tailoring of M016 is required for use by the WTP contractor as an implementing standard for design of structural steel for Seismic Category III SSCs.

No specific section

Load combinations for design of structural steel members utilize those identified in UBC 97, section 1612.3.

Justification: These load combinations represent the commercial requirements for allowable stress design of structural steel. Use of these load combinations will ensure compliance with the commercial design in accordance with the UBC.

No specific section

Seismic detailing requirements shall be in accordance wioth UBC 97, Chapter 22, Division V, section 2214, for moderate seismic risk structures.

Justification: The requirements contained in this section contain accepted industry practice for design of important commercial steel structures. Use of this section will ensure compliance with the commercial design in accordance with the UBC.

10.0 UBC 97, Uniform Building Code

Revision: 1997
Sponsoring Organization: International Conference of Building Officials

WTP Specific Tailoring

The following tailoring of UBC 97 is required for use by the WTP contractor as an implementing standard for design of reinforced concrete for Seismic Category III SSCs, as noted.

Division II Snow

Design for snow loads shall be in acordance with ASCE 7, *Minimum Design Loads for Buildings and Other Structures*, section 7.0, utilizing ground snow loads identified in Safety Criterion 4.1-4.

Justification: This approach to design of snow loads is an acceptable industry practice for evaluation of structures under snow loads. This code is more thorough in its consideration of these loads than the UBC methodology.

Division III Wind

Design for wind loads shall be in accordance with ASCE 7, *Minimum Design Loads for Buildings and Other Structures*, section 6.0, utilizing 3-second gust values identified in Safety Criterion 4.1-4.

Justification: This approach to design of wind loads is an acceptable industry practice for evaluation of structures under wind loads. This code is more thorough in its consideration of these loads than the UBC methodology.

Appendix D

Radiological Exposure Standards for the RPP-WTP Project

CONTENTS

{ TOC \o “1-3” \b “appD” }

1.0 Introduction and Purpose

This attachment to the SRD originally was issued as a stand-alone document (BNFL-5193-RES-01, Rev. 0, dated August 28, 1997). It has been incorporated into the SRD because it provides both background information and the basis for the radiological exposure standards reflected in the SRD Safety Criteria. In addition, it has been updated to reflect responses to DOE questions on the Standards Approval Package.

This document is the Radiation Exposure Standard for Workers Under Accident Conditions, which is a radiological safety deliverable. This document is used during the process hazards analysis (PHA) and accident analysis to ensure worker safety through identification of the need for accident prevention and mitigation features that provide worker protection against radiological and nuclear hazards. In this document, where unmodified reference is made to workers, it applies collectively to facility workers and collocated workers as defined in Sections 3.5.1 and 3.5.2 below.

The U.S. Department of Energy (DOE), in DOE/RL-96-0006, Revision 0, *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for TWRS Privatization Contractors*, (DOE-RL 1996), provides Table 1, "Dose Standards Above Normal Background". In Table 1 (referred to as DOE Table 1), there are entries labeled, "To be derived", for which the contractor is to propose specific exposure standards for both facility workers and collocated workers for the following events:

- **Unlikely Events**: events that are not expected but may occur during the lifetime of the facility in the range of frequency between 10^{-2}/yr and 10^{-4}/yr (between once in 100 years and once in 10,000 years)
- **Extremely Unlikely Events**: events that are not expected to occur during the lifetime of the facility but are postulated because their consequences would include the potential for the release of significant amounts of radioactive material. Extremely unlikely events are in the range of frequency between 10^{-4}/yr and 10^{-6}/yr (between once in 10,000 years and once in 1 million years).

This document provides the required exposure standards and the bases for their selection. In addition, this document presents the approach for complying with DOE Table 1. The individual elements of this approach, as shown in Table 2-1 of SRD Safety Criterion 2.0-1 (referred to as Table 2-1), are conservative based on the requirements of the contract and, as such, satisfy the contract. For completeness, this document also discusses, and presents in Table 2-1, public exposure standards and the assumed locations of the public, facility worker, and collocated worker for use in evaluation of accident consequences and normal radioactive material releases.

2.0 Exposure Standards for Facility and Collocated Workers

The four "To be derived" cells in DOE Table 1 have been completed by imposing a radiological exposure standard not to exceed 25 rem/event to the RPP-WTP facility workers or to collocated workers for either unlikely or extremely unlikely events.

The 25 rem/event exposure standard for both the facility and collocated workers for unlikely and extremely unlikely events corresponds to the once-in-a-lifetime accident or emergency exposure for radiation workers which, by recommendation of the National Committee on Radiation Protection (NCRP 1963), may be disregarded in the determination of their radiation exposure status. In addition, an exposure of 25 rem/event corresponds to a conditional probability of fatality of about 2×10^{-2}. For unlikely events (defined in Table 2-1 as having a maximum occurrence frequency of 10^{-2}/yr), this equates to a maximum increase in worker lifetime risk of premature death of only 2×10^{-4}, which is considerably less than the average accidental death risk for workers in some of the safest industries (i.e., retail and wholesale trade, manufacturing, and service [EPA 1991]).

Compliance with the 25 rem/event standard is established using qualitative methods supported, where necessary, by numerical analysis that may include the development of event trees and fault trees and/or the performance of consequence analyses. From this process, preventative and mitigative engineered and administrative controls are identified.

Use of qualitative methods is consistent with the American Institute of Chemical Engineers (AIChE) guidelines (AIChE 1992), U.S. Nuclear Regulatory Commission (NRC) guidance for the performance of integrated safety analysis for 10 *Code of Federal Regulations* (CFR) 70 special nuclear material licensees (NRC 1995a), as well as DOE-STD-3009 (DOE 1994) and DOE G 420.1-X (DOE 1995). Both DOE documents state the following:

> "Estimates of worker consequences for the purpose of a safety-significant SSC designation are not intended to require detailed analytical modeling. Considerations should be based on engineering judgement of possible effects and the potential added value of safety-significant SSC designation."

Because the primary purpose of the RPP-WTP Project facility and collocated worker exposure standards is to identify structures, systems, and components (SSC) required to protect these workers, the guidance cited above is both applicable and appropriate.

The principal approach for complying with the 25 rem/event worker exposure standard is the PHA. The PHA is a systematic, team-based review of the plant and treatment processes. The PHA identifies hazards and operability problems to a level of detail commensurate with the design detail available. Further hazard evaluation takes place in parallel with design development to ensure that safety continues to be built into the design process.

Having generated the list of hazards and hazardous situations, this list is subject to a further systematic team-based review where a binning process takes place. The binning process assigns postulated events to a certain hazard category and is essentially risk-based with categories of hazard defined according to a frequency/consequence matrix.

The 25 rem/event standard for unlikely or extremely unlikely events applies to events with frequencies less than 10^{-2}/yr. For those frequencies, the PHA process assigns serious and major hazardous situations as undesirable, acceptable with controls, or acceptable. For a hazardous situation to be “acceptable”, its consequences must be less than 25 rem. Where there is uncertainty as to where an event should be binned (i.e., assigning a hazard category), it is binned into a higher category to ensure that the accident analysis remains conservative.

The DOE-RU has provided a guidance document (DOE-RL 1997) to be used for review of the Radiation Exposure Standard for Workers Under Accident Conditions. This guidance document includes the worker accident risk goal and the accident risk goal of DOE/RL-96-0006.

The worker accident risk goal is stated in DOE/RL-96-0006 as, “The risk, to workers in the vicinity of the Contractor’s facility, of fatality from radiological exposure that might result from an accident should not be a significant contributor to the overall occupational risk of fatality to workers”.

DOE/RL-97-09 (DOE-RL 1997) describes approaches that can be taken to meet this goal. The simplest approach notes that the goal can be met when (a) a worker dose standard that does not exceed 100 rem is used for extremely unlikely events (10^{-4} to 10^{-6} probability range), and (b) a worker dose standard that does not exceed 10 rem is used for unlikely events (10^{-2} to 10^{-4} probability range). For the latter probability range, the 10-rem standard relies on the assumption that the probability of accidents is evenly distributed across the probability range.

Based on experience with similar plants, it is considered unlikely that the even distribution assumption will represent the actual situation for RPP-WTP. Furthermore, experience indicates that there will be relatively few accidents falling into this range, and that they will be distributed toward the low probability end of the range. Consequently, a value higher than 10 rem can be used for the worker accident standard for unlikely events.

As can be seen in Table 2-1, a value of 25 rem/event is selected as the worker accident standard for both unlikely and extremely unlikely events. Because this is over 10 rem for the 10^{-2} to 10^{-4} probability range, satisfaction of the worker accident risk goal needs to be demonstrated.

For the RPP-WTP, this goal is satisfied by calculating the risk of facility operation to the workers. This is a best estimate analysis based on realistic input and modeling assumptions. In performing this analysis, all structures, systems, and components capable of preventing or mitigating the event are considered. Estimates of system and component unavailabilities and unreliabilities consider failure to start and failure to run as well as maintenance-caused unavailabilities. Accident prevention and mitigation features are added to the design as necessary to satisfy the worker accident risk goal. Note 2 of Table 2-1 explicitly commits the RPP-WTP contractor to this risk evaluation process.

The accident risk goal is stated in DOE/RL-96-0006 as, "The risk, to an average individual in the vicinity of the Contractor's facility, of prompt fatalities that might result from an accident should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed." The DOE guidance document states that a radiation exposure standard of 100 rem/event would satisfy the accident risk goal. Because the RPP-WTP standard is 25 rem/event, the guidance document is satisfied.

In each of the four cells addressing accident exposure standards for workers and collocated workers in the unlikely and extremely unlikely events ranges, an ALARA accident limit is not specified. However, Note 3 of Table 2-1 states:

> "In addition to meeting the listed dose standards for accidents, the approach to accident mitigation is to evaluate accident consequences to ensure that the calculated exposures are far enough below standards to account for uncertainties in the analysis, and to provide for sufficient design margin and operational flexibility."

This approach provides an adequate level of safety. The following paragraphs should also be noted in support of this conclusion.

The accident analyses will show compliance with exposure standards for accidents. In addition, a defense-in-depth approach provides multiple levels of protection that ensure worker exposures from accidents will be significantly lower than calculated. This is a proven approach, considered to be effective at minimizing exposures to workers.

The approach to accident mitigation (as described in Note 3 of Table 2-1) is to examine accident consequences to ensure that calculated exposures are far enough below standards to account for uncertainties in the analysis and to provide sufficient design margin and operational flexibility. This approach is employed for all accidents (including both public and workers at all accident frequency levels) that can challenge the exposure standards, ensuring that accident exposures would be well below standards.

3.0 Development of the BNI Approach to Compliance with Table 1 of DOE/RL-96-0006

The overall approach to complying with DOE Table 1 is presented in this document. This approach takes the form of Table 2-1. The "To be derived" cells have been completed as discussed. The remaining cells of Table 2-1 are either identical or conservative with respect to DOE Table 1. The following sections discuss differences between DOE Table 1 and Table 2-1.

DOE Table 1 footnotes are not shown in Table 2-1. Section 2.1 of DOE/RL-96-0006 states that the footnotes refer only to the origin of the specific standards and, as such, are not considered contractual requirements unless included elsewhere in the contract.

3.1 Estimated Frequency of Occurrence

The second column of DOE Table 1, "Estimated Probability of Occurrence (P) (yr^{-1})," has been titled in Table 2-1, "Estimated Frequency of Occurrence (f) (yr^{-1})". In addition, the estimated frequency of occurrence for normal events of DOE Table 1 is redefined in Table 2-1 as any normal event regardless of frequency (nominally taken to be a frequency > 0.1/yr). The estimated frequency of anticipated events in DOE Table 1 is redefined as events with an annual frequency of occurrence of $10^{-2} < f\ 10^{-1}$.

With these changes, events routinely performed (e.g., melter replacement) are considered normal events rather than accidents, irrespective of frequency of occurrence. As normal events, the radiological assessment is subject to the more restrictive "per year" exposure standards rather than "per event" exposure standards. Consequently, these changes are conservative in comparison to DOE Table 1.

3.2 Normal Events/Public and Workers Exposure Standards

Clarifying descriptions have been included in the Normal Events/Public cell of Table 2-1 explaining that the second 100 mrem/yr standard applies to a member of the public entering the controlled area and the 25 mrem/yr standard is the public primary exposure standard for radioactive waste. The removal of DOE Table 1 footnotes (as noted above) necessitated the addition of these clarifying notes.

For the Normal Events/Worker and Normal Events/Collocated Worker cells of Table 2-1, the DOE Table 1 standard of 1.0 rem/yr ALARA design limit is replaced by a standard of 1.0 rem/yr ALARA design objective per 10 CFR 835, Section 1002(b). The corresponding worker standards for normal events in DOE Table 1 are tied to the ALARA design objectives of 10 CFR 835.1002(b) by the footnotes to DOE Table 1.

BNI has committed to full compliance with 10 CFR 835 in the SRD, and the other sections of 10 CFR 835.1002 provide adequate requirements to ensure routine worker exposures will be ALARA. In addition, a footnote, Note 1, is included in Table 2-1. This note states the following:

> "In addition to meeting the listed design objective of 10 CFR 835.1002(b), the inhalation of radioactive material by workers and collocated workers under normal conditions is kept ALARA through the control of airborne radioactivity as described in 10 CFR 835.1002(c)."

3.3 Anticipated Events/Worker and Collocated Worker Exposure Standards

References to as low as reasonably achievable (ALARA) standards have been removed for the Anticipated Events/Worker and Collocated Worker cells of Table 2-1. The ALARA design objective of 10 CFR 835, "Occupational Radiation Protection", is applied to normal events as shown in Table 2-1. However, with the redefinition in Table 2-1 of anticipated events as those events with an annual frequency of occurrence of $10^{-2} < f \leq 10^{-1}$, the ALARA objective no longer applies because anticipated events are not part of normal operation.

This change complies fully with Section 3.2, "Radiation Protection Objective", of DOE/RL-96-0006, which states the following:

> "Ensure that during normal operation radiation exposure within the facility and radiation exposure and environmental impact due to any release of radioactive material from the facility is kept as low as is reasonably achievable (ALARA) and within prescribed limits, and ensure mitigation of the extent of radiation exposure and environmental impact due to accidents."

This aspect of Table 2-1 also represents compliance with contractual requirements because footnote 3 of DOE Table 1 references 10 CFR 835.1002(b). This section, and 10 CFR 835.202 which it references, establishes design requirements for occupational exposures other than planned special exposures and emergency exposures. Administrative limits for planned special exposures and emergency exposures are addressed in 10 CFR 835.204 and 10 CFR 835.1302 and are complied with by the RPP-WTP.

Finally, to provide an adequate level of safety and to ensure that cost-effective safeguards affecting anticipated events are evaluated (and incorporated as appropriate) whenever the final calculated event consequence to a worker or collocated worker is 1 rem or more, the approach specifies a 1.0-rem/event design action threshold standard. In addition, a note is included in Table 2-1 to explain the application of the standard. This note (Note 4 to Table 2-1) states:

> "When a calculated accident exposure exceeds this threshold, then appropriate actions are taken. These include carrying out a less bounding (i.e., more realistic) evaluation to show that the accident consequences will be below the threshold or evaluating additional safeguards for cost-effectiveness and/or feasibility. This threshold is not a limit; it does not require the implementation of additional preventative or mitigative features if they are not both cost-effective and feasible."

3.4 Extremely Unlikely Events/Public Exposure Standard

A standard is included in the Extremely Unlikely Events/Public cell of Table 2-1 stating that a public exposure standard target value of 5 rem/event is applied to extremely unlikely events. This target value is based on the following:

- The philosophy is that the public should be protected by a lower exposure standard than a worker. This philosophy recognizes the fact that the worker has agreed to work on the Hanford Site and has received training for avoiding hazards and dealing with hazardous situations.
- A goal to facilitate transition to the NRC as the regulatory agency with jurisdiction over nuclear safety for DOE facilities. With the exception of a 25 rem/event guideline value of 10 CFR 100 for the establishment of the exclusion area and low population zone for commercial power reactors, the NRC has not established a public exposure standard that exceeds 5 rem/event. A public exposure standard of 5 rem/event is also included in proposed rulemaking for 10 CFR 70 (NRC 1995b), which further supports the Table 2-1 value.
- With the same 5 rem/event public exposure standard for both unlikely and extremely unlikely events, there is no need to bin accidents in one of these two event frequency categories for the purpose of establishing protection of public safety.

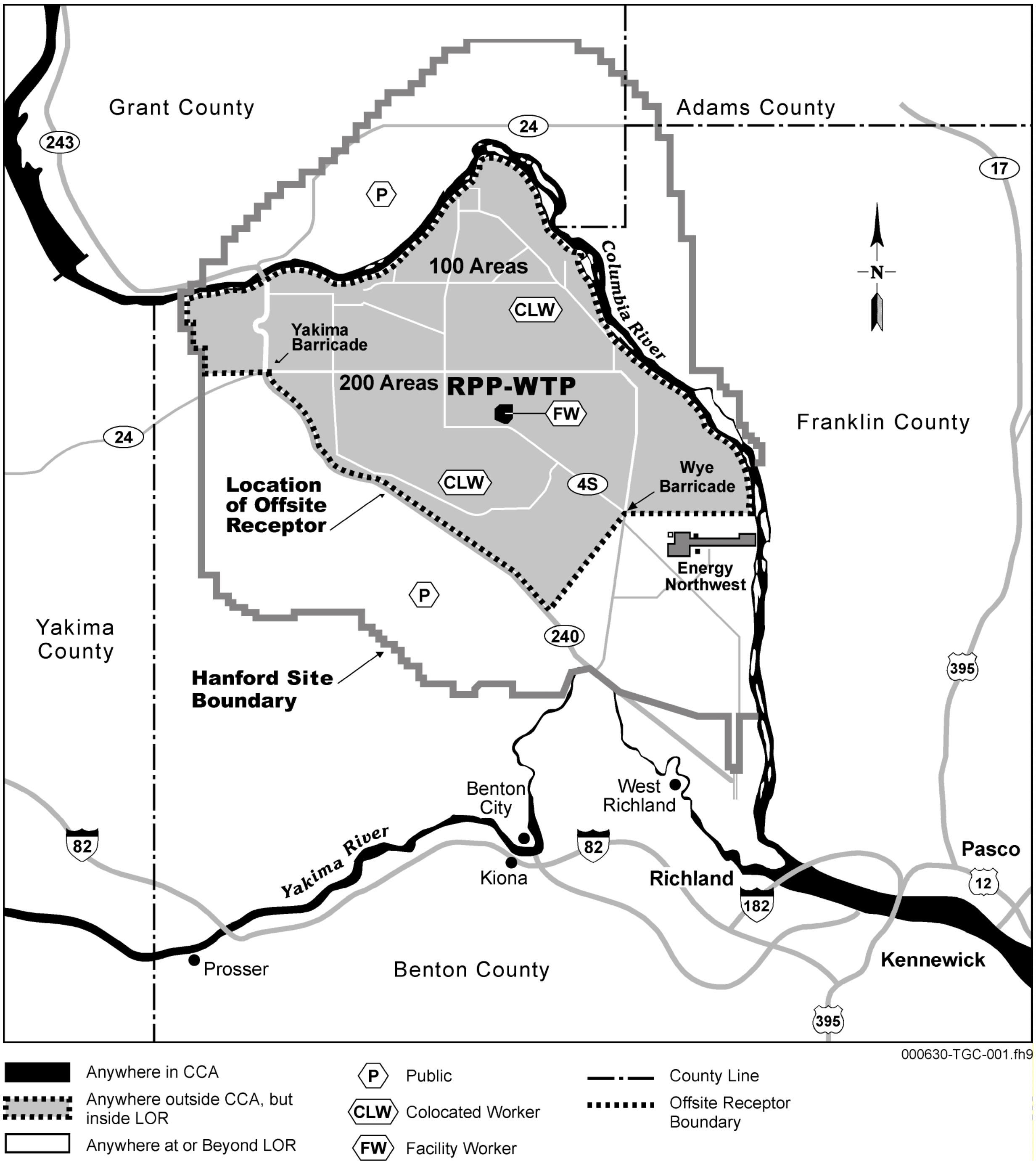
3.5 Location of Receptors

In Table 2-1, a new last row has been added to clarify in DOE Table 1 of DOE/RL-96-0006 the assumed location for the facility worker, the collocated worker, and the public, for the purpose of establishing compliance with the radiological standards of DOE Table 1. The bases for the receptor locations included in this row are provided below.

3.5.1 Facility Worker

The facility worker is located at the most limiting location within the RPP-WTP contractor-controlled area as defined in DOE/RL-96-0006, as shown in Figure D-1.

Figure D-1. Location of Facility and Collocated Workers.



Section 6.0, "Glossary", of DOE/RL-96-0006 defines the controlled area as the following:

> "The physical area enclosing the facility by a common perimeter (security fence). Access to this area can be controlled by the Contractor. The controlled area may include identified restricted areas."

The controlled area for RPP-WTP used to define the location of the facility worker, is that land within the RPP-WTP security fence.

3.5.2 Collocated Worker

Section 6.0, "Glossary", of DOE/RL-96-0006 defines the collocated worker as the following:

> "An individual within the Hanford Site, beyond the Contractor-controlled area, performing work for or in conjunction with DOE or utilizing other Hanford Site facilities."

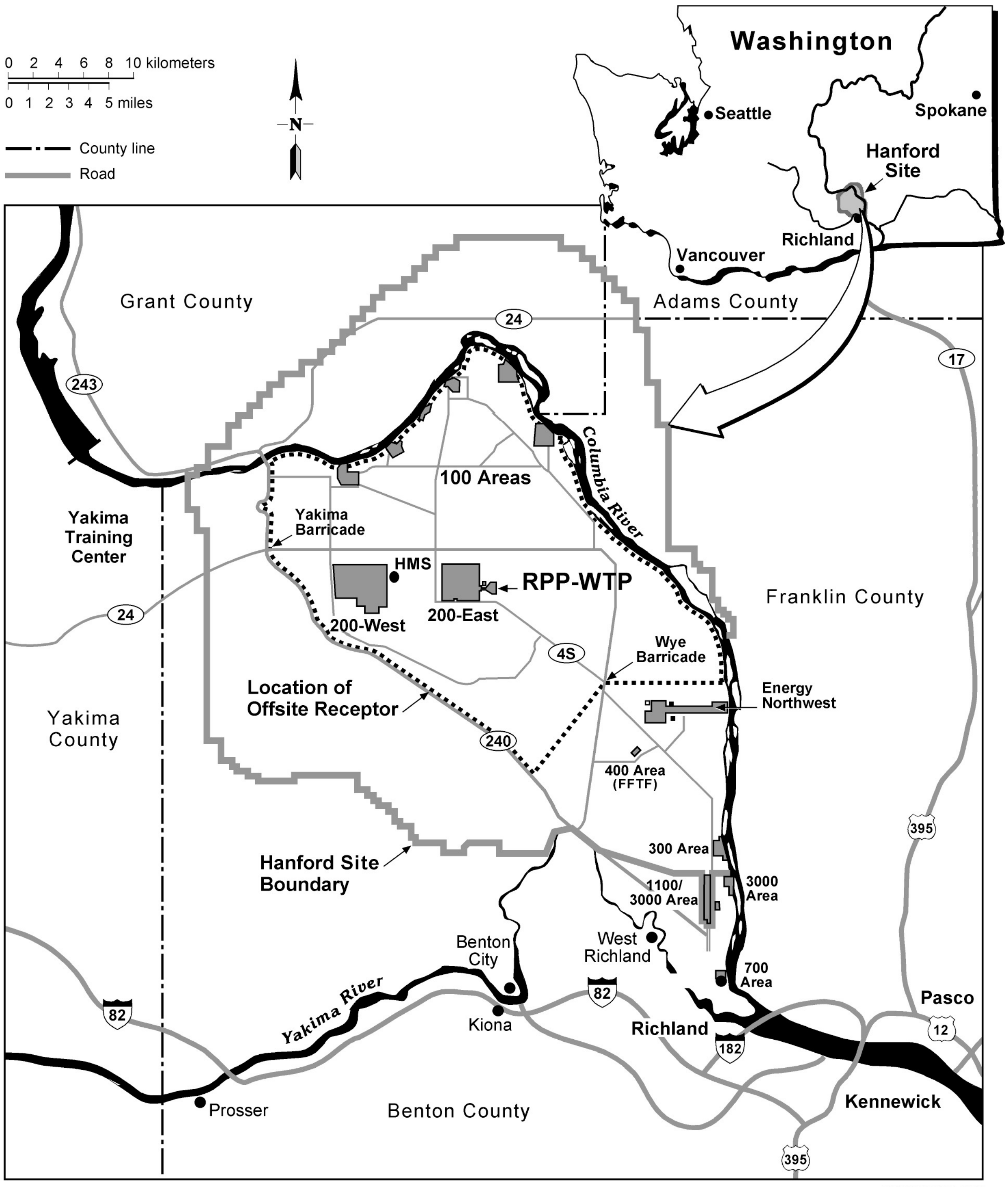
For evaluation of the RPP-WTP design to the exposure standards of DOE Table 1, the location of the collocated worker is either at the controlled area boundary or beyond that boundary if such a location results in higher exposure. For a ground-level release, the location of the collocated worker is considered no closer than 100 m from the release point.

3.5.3 Public

The location of the public (i.e., the offsite receptor) for the purpose of establishing compliance with the last column of DOE Table 1 of DOE/RL-96-0006, is established at the most limiting exposure location along the near bank of the Columbia River, Highway 240, and a southern boundary as shown in Figure D-2.

This area includes land for which it is reasonable to assume DOE will retain the right to control activities and limit access under accident conditions for the operating life of the RPP-WTP. Specifying the near river bank excludes the Columbia River for which DOE does not control activities (DOE-RL 1995). Specifying Highway 240 excludes the Arid Lands Ecology Reserve of which DOE might relinquish control during the operating life of the RPP-WTP. The southern boundary serves to exclude Energy Northwest's Columbia Generating Station, a commercial nuclear power plant (whose workers should be considered members of the public), and the Hanford Site 300, 400, and 1100 Areas. The 400 Area includes the Fast-Flux Test Facility.

Figure D-2. Boundary for Location of Offsite Receptor for the Purpose of Implementing DOE/RL-96-0006, Rev. 0, Table 1, Public Exposure Standard.



In footnotes 10 and 12, DOE Table 1 of DOE/RL-96-0006 makes reference to 10 CFR 72, "Licensing Requirements for the Independent Spent Fuel (ISFSI) and High Level Radioactive Waste," and 10 CFR 100, "Reactor Site Criteria," to relate to the public exposure standards for unlikely and extremely unlikely events. While the siting requirements and guidance of Parts 72 and 100 are not applicable to the RPP-WTP, the requirements for establishing the location of the offsite receptor in these two cited regulations are useful for locating the offsite receptor for a waste processing facility such as RPP-WTP. Section 72.106, "Controlled Area Boundary of an ISFSI or Monitored Retrievable Storage (MRS)", includes the following statements relative to the boundary to be assumed for the evaluation of radiological exposure to the public:

> "The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters."
>
> "The controlled area may be traversed by a highway, railroad or waterway, so long as appropriate and effective arrangements are made to control traffic and to protect public health and safety."

Title 10 CFR 100 establishes a guideline value of 25 rem for 2 hr at the exclusion area boundary. For the exclusion area, 10 CFR 100.3, "Definitions", states the following:

> "(a) *Exclusion area* means that area surrounding the reactor, in which the reactor licensee has the authority to determine all activities including exclusion or removal of personnel and property from the area. This area may be traversed by a highway, railroad, or waterway, provided these are not so close to the facility as to interfere with normal operations of the facility and provided appropriate and effective arrangements are made to control traffic on the highway, railroad, or waterway, in case of emergency, to protect the public health and safety. Residence within the exclusion area shall normally be prohibited. In any event, residents shall be subject to ready removal in case of necessity. Activities unrelated to operation of the reactor may be permitted in an exclusion area under appropriate limitations, provided that no significant hazards to the public health and safety will result."

As can be seen from the above excerpts, the assumed location for the offsite receptor for RPP-WTP is consistent with 10 CFR 72 and 10 CFR 100. In addition, the proposed southern boundary takes advantage of the road junction at the Wye barricade (Figure F-1) for control of access to the site during accident conditions.

4.0 References

10 CFR 70, "Domestic Licensing of Special Nuclear Material", *Code of Federal Regulations*, as amended

10 CFR 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste", *Code of Federal Regulations*, as amended

10 CFR 100, "Reactor Site Criteria", *Code of Federal Regulations*, as amended

10 CFR 835, "Subpart C - Standards for Internal and External Exposure", *Code of Federal Regulations*, as amended

AIChE, 1992, *Guidelines for Hazards Evaluation Procedures, Second Edition with Worked Examples*, Center for Chemical Process Safety, American Institute of Chemical Engineers, New York, New York

DOE 1994, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Safety Analysis Reports*, DOE-STD-3009-94, U.S. Department of Energy, Washington, D.C.

DOE 1995, *Implementation Guide for Nonreactor Nuclear Safety Design Criteria and Explosives Safety Criteria*, DOE G 420.1-X, Revision G, U.S. Department of Energy, Washington, D.C.

DOE-RL 1995, *Clarification of Hanford Site Boundaries for Current and Future Use in Safety Analysis*, letter Walter B. Scott, DOE-RL to Contractors, dated September 26, 1995, U.S. Department of Energy, Richland Operations Office, Richland, Washington

DOE-RL 1996, *Top-Level Radiological, Nuclear, and Process Safety Standards and Principles for TWRS Privatization Contractors*, DOE/RL-96-0006, Revision 0, U.S. Department of Energy, Richland Operations Office, Richland Washington

DOE-RL 1997, *Guidance for Review of TWRS Privatization Contractor Radiation Exposure Standards for Workers*, DOE/RL-97-09, U.S. Department of Energy, Richland Operations Office, Richland Washington

EPA 1991, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, U.S. Environmental Protection Agency, Washington, D.C.

NCRP 1963, *Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure*, Handbook 69, Addendum 1, National Bureau of Standards, Washington, D.C.

NRC 1995a, *Integrated Safety Analysis Guidance Document*, NUREG-1513, Draft, U.S. Nuclear Regulatory Commission, Washington, D.C.

NRC 1995b, *Preliminary Working Draft of Revision of 10 CFR 70 Updated*, 4/05/95, provided at the NRC public meeting of May 2, 1995, U.S. Nuclear Regulatory Commission, Washington, D.C.

Appendix E

Reliability, Availability, Maintainability, and Inspectability (RAMI)

To ensure that the facility meets operational requirements, it is necessary to address issues associated with reliability, availability, maintainability, and inspectability.

Reliability is used as a measure of the ability of an item or system to complete a task, and it is normally expressed as a probability of failure. Reliability is designed in through the use of appropriate design techniques and control of the mode of operation and the environment. Design techniques to be used vary because they are dependent on the specific item or system and the task to be performed. Their purpose is to optimize reliability by the following:

1) Use of proven materials and components
2) Design simplicity
3) Testability
4) Control of manufacturing standards
5) Control of operational mode (e.g., prevention of misuse and overloads)
6) Control of environment (e.g., protection against corrosion and vibration)

Consistent with the RPP-WTP process for tailoring hazard controls using the potential radiological and chemical consequences of individual events, reliability is assigned to SSCs based upon the importance of the SSC to the prevention or mitigation of accidents. The significance of accident prevention and mitigation is determined by the severity of the accident to workers or the public. To implement this tailoring in a clear, consistent, and defensible manner, an Implementing Standard for Safety Standards and Requirements Identification was developed. This Implementing Standard includes a Severity Level ranking system which provides the hazard assessment and control teams with a defined way to categorize the potential severity of those events that can result in radiological or hazardous exposure to the workers or the public. The Implementing Standard provides the means by which the hazard assessment and control teams establish target reliabilities for SSCs.

Availability is a measure of the degree to which an item or system is in an operable condition. It is expressed quantitatively as the ratio of the mean time between failures to the sum of the mean time between failures and the mean time to repair. System availability is calculated to determine the potential for downtime. In this way, systems are identified that contribute to decreased availability. Required availability is achieved by specifying additional systems or increasing reliability of existing systems.

Maintainability is a measure of the ability to restore a failed item or system to an operable condition in a specified time. Maintainability is designed into the facility and processes through use of appropriate design techniques, (e.g., the use of specially designed, remotely removable, and replaceable pumps and valves in process systems, and the placement of active pumps or valves within shielded accessible areas equipped with appropriate decontamination facilities that allow hands-on maintenance activities) and logistic support (e.g., scheduling and procedures). Benefits of these design techniques are that they simplify maintenance operations in high radiation areas and remove high maintenance equipment from high radiation areas. Testability of Safety Design Class systems and components is facilitated by such features as redundancy that allow for a system or component to be removed from service for maintenance or testing without loss of safety protection.

Inspectability is the measure of the ease with which items or systems can be inspected for preventative maintenance or assessment of condition. Inspectability is used to monitor facility items in order to maintain their reliability. Inspectability of facility items can be designed in by the use of shielded access areas (as above, to reduce radiation exposure) for active equipment or the provision of monitoring equipment (e.g., material coupons for determining vessel corrosion rates, and in-cell cameras).

During the design phase, the RPP-WTP facility and processes are evaluated for reliability, availability, maintainability, and inspectability. A number of validated modeling techniques (computer codes, mathematical modeling, failure modes, and effects analysis) for determining reliability and availability of the facility and processes are used. These are used to identify those facility and process areas that are sensitive with respect to influencing overall facility and process performance. Optimum reliability is established by the use of appropriate standards and quality control. The determination of maintenance and inspection needs is based on facility and process reliability requirements. It is a mixture of process optimization, provision of appropriate design features to aid preventative and scheduled maintenance and inspection, and the development of maintenance and inspection programs (administrative and procedural controls) whose objectives among other things, are to facilitate these activities. Reliability targets are assigned to SSCs only when a quantitative value has been credited for the reliability of an SSC in safety analysis.

Appendix F

Ad Hoc Implementing Standard for Deactivation and Decommissioning Planning

CONTENTS

1.0 Introduction

All elements of the RPP-WTP safety approach are applied to the deactivation phase of the project. In addition, the RPP-WTP will incorporate design provisions to facilitate deactivation and final decommissioning as described in the implementing standard DOE G 441.1-2, *Occupational ALARA Program Guide*, for SRD Criterion 8.0 - 2. These provisions will reduce radiation exposure to Hanford Site personnel and the public during and following deactivation and decommissioning activities and minimize the quantity of radioactive waste generated during deactivation. The purpose of this standard is to define the attributes that must be addressed during the preparation of the deactivation plan to protect both the Hanford Site personnel and the public both during and after the deactivation stage of the project.

2.0 Plan Preparation

A deactivation plan will be prepared prior to construction of the RPP-WTP. The deactivation plan will provide details on how the following activities will be accomplished to achieve a deactivated status for the facility.

1) Verification of the completion of the facility deactivation end point. The term facility deactivation end point refers to the set of conditions that comprise the completion of facility deactivation i.e., radiological, structural, equipment, and documentation. These general end points will be defined in the deactivation plan and a requirement made to determine specific end points. When these end point criteria are met the facility will be in a safe state that can be economically monitored and maintained until final decommissioning.
2) Documentation of the regulatory status, conditions, and inventories of remaining radioactive and hazardous materials and health and safety requirements. After facility construction but before deactivation commences, the deactivation plan will require a hazard evaluation for radiological, nuclear, and process safety be carried out. Safety standards and requirements will be identified to implement the controls to protect against the facility hazards.
3) Identification of the facilities, structures, support systems, and surveillance systems to provide for confinement and monitoring of the remaining contamination, radiation, and other potential hazards. After facility construction but before deactivation commences, the plan will be expanded to describe the activities required to maintain the operability of critical equipment and to maintain the structural integrity of the deactivated facility. It will identify modification requirements to systems for the above purposes.
4) Posting and securing of the facility. After facility construction but before deactivation commences, the plan will be expanded to identify the radiological controls required for the deactivated facility, which will include posting of radiological areas. The need for other safety postings will also be identified.

5) Removal of packaged special nuclear materials and other packaged radiological and chemical materials.
6) Confirmation that security systems and procedures are adequate and in place to prevent unauthorized entry.
7) Waste minimization during the deactivation process.

3.0 Summary

The above requirements for the deactivation plan in combination with measures taken at the design stage of the project will protect the Hanford Site personnel and the public both during and following the deactivation activities.

4.0 Definitions

Deactivation - Placing the facility in stable and known conditions, identifying hazards, eliminating or mitigating hazards, and transferring programmatic and financial responsibilities from the operating program to the disposition program. Surveillance and maintenance continues to assure public, environment, and worker safety. The facility is in a safe storage mode, with ongoing, low levels of surveillance and maintenance. The general intent is that the facility be unoccupied and locked except for periodic inspections. Radioactive and hazardous materials may remain in the facility and are subject to ongoing regulatory oversight. (DOE/EM-0318, *Facility Deactivation Guide -- Methods and Practices Handbook*, December 1996)

Decommissioning - The process of removing a facility from operation, followed by decontamination, entombment, dismantlement, or conversion to another use. (DOE G 430.1-1A, *Life Cycle Asset Management*)

Decontamination - The reduction or removal of contaminating radioactive material from a structure, area, object or person. Decontamination may be accomplished by (1) treating the surface to remove or decrease the contamination, (2) letting the material stand so that the radioactivity is decreased as a result of natural decay, and (3) covering the contamination to shield or attenuate the radiation emitted. (Health Physics and Radiation Health Handbook, Revised Edition, Bernard Shleien, 1992)

End Point - Specifying and achieving end points is a systematic, engineering way of proceeding from an existing condition to a stated desired final set of conditions in which the facility is safe and can be economically monitored and maintained. (DOE/EM-0318, *Facility Deactivation Guide - Methods and Practices Handbook*, December 1996)

Appendix G

Ad Hoc Implementing Standard for Safety Analysis Reports[CJD5]

CONTENTS

1.0 Introduction

The purpose of this Implementing Standard is to define the format and content for RPP-WTP safety analysis reports (SARs).

Section 2.0 provides the definitions important to this Implementing Standard. Section 3.0 defines the process for development, review, and approval.

2.0 Definitions

For the definitions of the following terms, see the reference provided.

Safety Analysis Report (SAR) (DOE/RL-96-0006 [DOE-RL 1998b])

3.0 Process

3.1 Safety Analysis Report Preparation

The River Protection Project Waste Treatment Plant (RPP-WTP) SARs document the safety analyses for the facility to demonstrate that it can be safety operated, maintained, and shut down.

The SARs shall be prepared in accordance with the requirements of:

1) DOE/RL-96-0003, *DOE Regulatory Process for Radiological, Nuclear, and Process Safety for TWRS Privatization Contractors* (DOE-RL 1998a), Sections 4.3.2 and 4.3.3, both titled “Contractor Input”
2) Contract Table S7-1, “Radiological, Nuclear, and Process Safety Deliverables”
3) *Safety Requirements Document Volume II* (SRD) (BNI 2001), Safety Criterion 9.1-2

The content of the Preliminary Safety Analysis Report (PSAR) and the Final Safety Analysis Report (FSAR) are developed using the guidance provided in the Nuclear Regulatory Commission’s 1995 draft revision to Regulatory Guide 3.52, *Standard Format and Content for Heatlh and Safety Sections of License Applications for Fuel Cycle Facilities* (NRC 1995). The content of the SARs is tailored to the nature of the RPP-WTP relative to the hazards and hazardous situations identified by the process hazards analysis. Planned deviations from the content guidance of draft Regulatory Guide 3.52 are identified in Table G-1.

The Table of Contents for the safety analysis reports follows Table G-1. The safety analysis report will not be submitted to the regulator until all major safety issues have been resolved and other safety issues have been scheduled for completion. The FSAR should identify significant changes made in the facility design and plans for operation from what was presented in the PSAR. The FSAR, in addition to including facility and process drawings, should also include fabrication and construction specifications important to the safety analysis of the facility.

Table G-1 Deviations from the Safety Analysis Report Content Guidance of Regulatory Guide 3.52 [1]

Chapters	Addition or Subtraction	Basis
1.3 Site Description	Regulatory Guide (RG 3.52) suggests that Section 1.3 summarize information used in preparing the Environmental Report. Specific information is referenced, but not duplicated in the safety analysis report (SAR).	The Environmental Report provides this information.
1.3.2 Demography and Land Use	The population distribution as a function of distance and direction is not to be provided. The distances to nearby population centers are provided.	There are no residences on the Hanford Site and the nearby population is low.
3.3 Quality Assurance	Section 3.3.4, "Quality Assurance Program Description" addresses the 10 criteria of 10 CFR 830 Subpart A, "Quality Assurance Requirements" in lieu of the 18 criteria listed in RG 3.52.	By contract compliance to the 10 CFR 800 series of nuclear safety requirements is required. This includes compliance to 10 CFR 830 Subpart A, "Quality Assurance Requirements". The differences in the criteria to be addressed are not significant because the quality assurance programs are based on consensus standards.
3.5 Human Factors	RG 3.52 states that a formal human factors program is not required if the facility has no requirement for safety-class actions. Human factors are considered in the Preliminary Safety Analysis Report (PSAR) independent of whether or not human actions are required for protection of the public or workers.	The requirements of DOE/RL-96-0006 (DOE-RL 1998a), Section 4.2.6, "Human Factors", extend beyond consideration of human factors as related to actions taken to protect the public. Final Safety Analysis Report (FSAR) Section 3.5 documents how compliance to contract Section 4.2.6 is achieved.
3.10 Testing Program and Preoperational Safety Review	This section is added to address the initial and commissioning testing programs.	Addition of this section facilitates documentation of compliance to DOE/RL-96-0006 (DOE-RL 1998b), Section 4.2.8, "Pre-Operational Testing", and Section 5.2.6, "Pre-Startup Safety Review", and DOE/RL-96-0003 (DOE-RL 1998a), Section 4.3.2, "Contractor Input", item 13.
3.11 Operational Practices	This section is to added to address such conduct of operations considerations as shift routine and turnover, control area activities, communications, control of on-shift training, control of equipment and system status, lockout and tagout, independent verification of equipment status, logkeeping, and operational aids postings.	These items are discussed to address what is normally considered conduct of operations.

Table G-1 Deviations from the Safety Analysis Report Content Guidance of Regulatory Guide 3.52 [1]

Chapters	Addition or Subtraction	Basis
4.7 Results of the Integrated Safety Assessment	The results for unmitigated accidents are compared to the radiological standards discussed in Integrated Safety Management Plan (ISMP) Section 1.2, "Detailed Description of the Safety Approach" rather than to 10 CFR 20, "Standards for Protection Against Radiation". A full assessment of the hazardous situations that might present themselves during facility operation is provided. This includes estimates of radiological and chemical releases for this range of events. Additional details are provided on the methodology used for consequence analysis, bounding conditions, input assumptions, and accident sequences.	The standards provided in RG 3.52 were derived from 10 CFR 20, "Standards for Protection Against Radiation", which is applicable to normal operation. The nature of the accidents for the RPP-WTP requires more discussion of consequence analysis than that required of fuel fabrication facilities.
4.8 Controls for Prevention and Mitigation of Accidents	This section identifies the specific safeguards selected for protection of the facility workers, as well as safeguards selected for protection of the public and collocated workers.	The nature of the accidents for the RPP-WTP requires more discussion of consequence analysis than that required for fuel fabrication facilities.
5.0 Radiation Safety	Chapter 5.0 provides the upper-level statutory standards and program policies that ensure the radiological safety of employees, visitors, and onsite members of the public. Deviations from RG 3.52 are as follows: 1) As an U.S. Nuclear Regulatory Commission (NRC) document, RG 3.52 references and specifies applicable portions of 10 CFR 20. Because 10 CFR 835 is the radiation safety regulation for the RPP-WTP, the focus of this section is on 10 CFR 835. 2) The implementation-level standards and guidance documents referenced in RG 3.52 is being incorporated into the Radiation Protection Plan (RPP).	Compliance with 10 CFR 835 is a requirement of the contract. The RPP required by 10 CFR 835 is required to include some of the information required of RG 3.52. There is no need to present this information in two documents.
5.1 As Low As Reasonably Achievable (ALARA) Policy and Program	RG 3.52 states that Regulatory Guide 8.10, Revision 1R (*Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable)* should be used in the development of the ALARA program. DOE guidance such as DOE G 441.1-2, *Occupational ALARA Program Guide* will also be used to develop the RPP-WTP ALARA program for normal operation.	DOE practices have proven to be successful for facilities similar to the RPP-WTP.

Table G-1 Deviations from the Safety Analysis Report Content Guidance of Regulatory Guide 3.52 [1]

Chapters	Addition or Subtraction	Basis
5.3 Radiological Safety Standards	Section 5.3 is added to provide the radiation standards by which the program operates. The standards specifically identify regulatory exposure standards, administrative exposure control levels, and other key standards of the radiation protection program.	The contract requires compliance to the 10 CFR 800 series of nuclear safety requirements. This includes compliance to 10 CFR 835, "Occupational Radiation Protection". Section 5.3 documents the compliance to the exposure standards of those regulations that have been promulgated.
5.8 External Exposure (renumbered 5.9 from RG 3.52)	By RG 3.52, the applicant is expected to participated in the National Voluntary Laboratory Accreditation Program (NVLAP) external dosimetry. Section 5.8 allows for participation in either the NVLAP or U.S. Department of Energy (DOE) Laboratory Accreditation Program (DOELAP) accreditation programs.	The option of participating in either the NVLAP or the DOELAP provides maximum flexibility and equivalent dosimetry program quality
5.14 Radioactive Waste Management	RG 3.52 does not require a discussion of waste management systems.	Section 5.14 is added to the SARs as the Process Hazards Analysis (PHA) completed for the RPP-WTP have identified hazards and hazardous situations with the waste management features of the facility. It is a requirement of DOE/RL-96-0003 (DOE-RL 1998a), Section 4.1.2, "Contractor Input", that deliverables be tailored to the nature and level of hazards associated with its waste processing activities.
Appendix 5A Radiation Protection Program Outline	This appendix is added to address compliance to 10 CFR 835.	The contract requires compliance to the 10 CFR 800 series of nuclear safety requirements. This includes compliance to 10 CFR 835, "Occupational Radiation Protection".
Appendix 5B Environmental Radiation Protection Program Outline	This appendix is added to address compliance to the requirements of the Environmental Protection Agency (EPA) and Washington State laws and regulations.	The contract requires submittal of an outline for the environmental radiological protection plan.
Chapter 6.0 Nuclear Criticality Safety	The methodology for criticality analyses is provided in the SARs to the extent the need to perform criticality calculation is found to be appropriate. The RPP-WTP SARs provide fewer details and commitments compared to fuel fabrication facilities relative to: 1) Nuclear criticality safety organization (Section 6.2.1) 2) Criticality training (Section 6.2.5) 3) Specific maintenance and quality assurance provisions for criticality prevention (Sections 6.2.3 and 6.2.4) 4) Audits and inspection (Section 6.2.6)	RG 3.52 focuses heavily on accidental criticality which is a more significant concern for fuel fabrication facilities which have a much higher inventory and concentrations of fissile material than the RPP-WTP. See ISMP Section 3.8, "Criticality Safety", for additional information.

Table G-1 Deviations from the Safety Analysis Report Content Guidance of Regulatory Guide 3.52 [1]

Chapters	Addition or Subtraction	Basis
7.4 “Hazardous Waste Management”	Section 7.4 of the RPP-WTP SARs address all chemical inventories that are identified by the PHA as representing a significant hazard.	By Section 4.2.2, “Contractor Input”, of DOE/RL-96-0003 (DOE-RL 1998a), the Initial Safety Analysis Report (ISAR) is to address process safety as well as radiological and nuclear safety. The need to address all aspects of chemical safety is also a NRC requirement of RG 3.52, Section 7.4, and NUREG-1513, “Integrated Safety Analysis Guidance Document”, (draft) (NRC 1994). The NUREG-1513 definition of “integrated” provided in Section 2.1, “Definition”, makes reference to chemical safety. Specific guidance for chemical safety is provided in Section 2.6.2, “Process Safety Information”, of the NUREG-1513.
10.0 Environmental Protection	This chapter references the Environmental Report	Protection of the environment is addressed in a separate document.
11.0 Deactivation and Decommissioning	This chapter addresses design and operational provisions considered to facilitate deactivation and decommissioning. It does not address the financial considerations for decommissioning.	The scope of the contract (DOE-ORP 2000) is limited to design support for deactivation.

1. Standard Format and Content for the Health and Safety Sections of License Applications for Fuel Cycle Facilities, Regulatory Guide 3.52, Revision 2, draft, U.S. Nuclear Regulatory Commission, Washington D.C. (NRC 1995).

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
1.1.1 Facility Description	A description of the facility design is provided in sufficient detail to demonstrate the facility design and construction requirements of the Safety Requirements Document (SRD). The details are also sufficient to support an understanding of the safety analysis provided in section 4.2, “Facility Description”.	This section updates the general description of the facility design.
1.1.2 Process Description	This section describes the process design in sufficient detail to demonstrate the system and component design and fabrication requirements of the SRD are satisfied. Details on the process design sufficient to support an understanding of the safety analysis are provided in section 4.3, “Process Description”.	This section updates the general description of the process design.
1.2 Institutional Information	This section provides the information required by RG 3.52, draft (NRC 1995a).	This section updates any changes in the institutional information provided in the PSAR.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
1.3 Site Description	A description of the site land use, meteorology, hydrology, geology, and seismology is provided.	This section address any existing or planned changes in land use from that provided in the PSAR. The FSAR provides any new meteorology, hydrology, geology, and seismology data made available. However, the level of detail provided for these subject areas is not significantly different between the two SARs. The FSAR summarizes data obtained during the Facility excavation that confirms the adequacy of the design. This includes the results of field and laboratory investigation of soil properties.
2.1 Organization and Administration	The Project organizational charts with a focus on the design and construction management organizations are provided. An organization chart for the operational phase is also presented. More definitive information on the roles, responsibilities, and interfaces for project management, engineering, construction management, inspections, procurement, quality assurance, records management, and nuclear safety functions is included. Section 2.1 also provides the criteria to determine minimum staffing requirements. A summary of procedures to be developed to implement the regulatory requirements addressed in this section is presented.	The section contains an update to the organizational structure of Project with a focus on operational and operational support organizations. This section also includes: 1 Title of each position that is important to public and worker safety and reporting relationship 2 Description defining qualifications, responsibilities, and authorities for each position related to safety 3 Organizational charts of the line organization and safety organization 4 Title of the individual delegated overall responsibility for the safety programs who has the authority to shut down operations if they appear to be unsafe, including independence of this authority from operational constraints 5 Lines of responsibility and authority for safety 6 Lines of communication and interfaces between organizations inside the facility 7 Availability of personnel within the safety organization to carry out the assigned function Specific information on procedure development and minimum staffing requirements is provided.
2.2 Safety Committees	Information on responsibilities, authorities, and proposed charters of safety committees, and oversight groups is provided.	This section updates information on safety committees, and oversight groups that are established following issuance of the PSAR and addresses any new safety committees that have been established.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
3.1 Configuration Management	This section contains specific information on: 1 Content and reference to procedures used to maintain effective configuration management of the RPP-WTP 2 Scope of identified systems, structures, and components (SSCs) and their relationship to the contents of Chapter 4.0, "Integrated Safety Analysis" 3 Description of the design information package contents to be provided to the safety analysts 4 Change control system specifics, including identification, technical and management reviews, documentation, and implementation 5 Specific physical configuration assessment, and periodic equipment performance monitoring 6 Design, installation, and testing of facility modifications 7 Revision of operating, test, calibration, surveillance, and maintenance procedures and drawings 8 Selection and control of replacement parts 9 Description of how the RPP-WTP design requirements and design basis were established and documented A summary of procedures developed to implement the regulatory requirements addressed in this section 3.1 is presented. This section also includes a draft of the unreviewed safety question process.	Specific information on the content of procedures and training developed is provided. The final unreviewed safety question process is provided.
3.2 Maintenance	A list of Safety Design Class and Safety Design Significant SSCs is provided. The maintenance implementation plan is described to such a level that maintenance philosophy and approach are evident.	The FSAR may modify the list of SSCs actions to be addressed based on safety analysis of the final design. Specific information on procedures and training developed to implement the requirements of section 3.2 is provided. In addition, the elements of the finalized maintenance implementation plan is described. Also discussed is the application of information obtained from demonstration testing and commissioning programs to the maintenance program (the latter by FSAR amendment after initial submittal).

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
3.3 Quality Assurance	Information related to the roles, responsibilities, and interfaces for project management, engineering, construction management, inspections, procurement, quality assurance, records management, and nuclear and process safety functions is provided. Included is the organizational structures of the quality assurance organization. The PSAR describes the quality assurance requirements of SSCs. Requirements for procedures to implement the regulatory requirements is presented.	For the FSAR, this section focuses on the quality assurance program for the operating RPP-WTP. Specific information on procedures and training developed to implement the requirements of section 3.3 is provided.
3.4 Training and Qualification	A description of the performance-based training program for operational and support personnel, including a detailed description of the training development process, is provided. The administrative process to be applied to training activities is described to a level such that the elements of the program and management's commitment to training is evident.	Details on the training and qualification program are provided. Also discussed is the application of information obtained from demonstration testing and commissioning programs (the latter by FSAR amendment after initial submittal).
3.5 Human Factors	This section documents the criteria by which human factors are considered in the facility design and operation.	This section states how human error in facility operations was taken into account in the design by facilitating correct decisions by operators and inhibiting wrong decisions. Consideration given in the design to detecting and correcting or compensating for errors is discussed.
3.6 Audits and Assessments	Information on the performance of audits and assessments is incorporated into this section.	This section is focused on audits and assessments performed during RPP-WTP operation. Specific information on procedures and training developed to implement the requirements of this section is provided.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
3.7 Incident Investigation	This section includes the following: 1 Provisions for establishing investigating teams 2 Functions, responsibilities, and scope of authority of investigating teams 3 Qualifications of internal and/or external investigators on investigating teams 4 A description of the procedures to ensure prompt investigation of an incident 5 Policy directives that the investigative process and the investigating team be independent of line management and that participants be assured of no retribution from participating in investigations 6 The approach proposed to determine the root cause(s) of incidents to ensure that the process is reasonable, systematic, and structured 7 Methods to ensure that corrective actions to resolve findings from incident investigations are tracked to completion 8 Identification and application of lessons learned 9 Specific reporting criteria for incident reporting during the construction phase. A summary of procedures developed to implement the regulatory requirements addressed in section 3.7 is presented.	Specific information on procedures and training developed to implement the requirements is provided. Included are specific reporting criteria for incident reporting during the operations phase.
3.8 Records Management	This section contains the organization structure and a description of the records management system, including authorities, responsibilities, and qualifications of personnel managing Environmental, Safety, and Health (ES&H) records. A summary of procedures developed to implement the regulatory requirements contained in section 3.8 is presented.	Specific information on procedures and training developed to implement the requirements is provided.
3.9 Procedures	A description of the administrative controls to ensure that work is performed in accordance with established technical standards and using approved instructions and procedures is provided.	This section describes the detailed processes of selecting activities requiring operating, emergency, and support procedures; preparing procedures; verifying and validating procedures; and reviewing and approving procedures. In addition, the program to administratively control procedures and their use is described in detail.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
3.10 Testing Program and Preoperational Safety Review	This section describes the analysis used to identify and define pre-operational and commissioning tests and describes tests required to ensure compliance to safety specifications. The testing program and controls are described to a level such that the testing philosophy and approach are evident. The prestart safety review approach is described to a level such that the areas to be evaluated and the evaluation approach are evident.	This section may modify the list of required safety improvement program and commissioning tests based on safety analysis of the final design. In addition, the administrative and program controls applicable to the test program are described in full.
3.11 Operational Practices	A description is provided of operational practices influenced by design details (i.e., communications systems, operational hazards associated with systems and hardware, and control area arrangements).	A description is provided of the operational practices influenced by the final design. In addition, final descriptions are provided on controls and administration of operational practices.
4.0 Integrated Safety Analysis	The methodology for hazards identification and accident analyses is described. The accident consequence analyses include margins in assumptions, boundary conditions, modeling and comparisons to acceptance criteria, as appropriate, to account for uncertainties in the design and plans for operation. Section 4.7 addresses the relationship of these uncertainties to the need to provide sufficient information in the construction authorization package to allow for issuance of the construction authorization.	Assumption used the PSAR to account for uncertainties in the design and plans for operations are removed from the FSAR analysis to the extent that these uncertainties have been resolved.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
4.2 Facility Description	In addition to providing a general description of the facility, this section discusses the basic civil/structural criteria to be applied to the design. For those structures classified as Safety Design Class, this includes the following: 1 Design codes, standards, and specifications 2 Loading criteria and load combinations 3 Design and analysis methodology 4 Structural acceptance criteria 5 Criteria for identifying testing and in-service inspection requirements 6 Material specifications 7 Special construction features This section also discusses: 1 Assumed soil properties 2 Excavation, backfill, and recompaction criteria 3 Assumed bearing capacity of the soil and the safety factor applied to this capacity 4 Expected static and dynamic building total and differential settlements. Less detail is provided for Safety Design Significant structures. Section 4.2 gives specific attention to those structures classified in Section 4.8 as Safety Design Class. Structures located away from the buildings containing significant hazards and that have no relationship to nuclear or process safety are briefly described (e.g., structural design, and the contents and functions of the building) and identified on a plot plan.	The FSAR updates the facility description and basic civil/structural criteria provided in the PSAR. It follows with discussions of the results of the application of these criteria to specific features of the facility. Examples are as follows: 1 The confirmation of soil properties obtained during excavation 2 A table providing the building total and differential settlement data obtained 3 Derived soil damping values 4 The results of the soil/structure analysis 5 Developed floor response spectra and time histories 6 A list of moderate and high energy systems 7 A list of specific missile and jet impingement sources, targets, and barriers provided. Also provided are updated plan and section drawings for structures classified as Important-to-Safety. These drawings show the basic floor arrangements, location of major systems and equipment, and basic building dimensions. For those structures classified as Safety Design Class, the drawings also show key structural elements, such as panel and floor reinforcements, cell liners, leak chases, major equipment anchors, and the use of masonry walls.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
4.3 Process Description	The description of process systems includes process flow diagrams for the major systems with instrumentation, sample points, and control features noted to the extent they have been developed. Heat loads are provided for heat transfer systems important to the safety analysis. Design features and parameters important to Section 4.7, "Results of the Integrated Safety Assessment", are provided. This section contains the following additional detail for each system classified as Safety Design Class: 1 The specified safety function(s) with reference to PSAR Section 4.7 for the basis 2 The design basis to be applied in the development of the system design 3 Design margins to be applied 4 The criteria to be used for the development of material specifications 5 Criteria to be used to determine design limits (such as pressure and temperature) 6 Criteria to be used to identify the need for instrumentation to monitor process conditions and the design criteria for such instrumentation (e.g., application of the single-failure criterion, and testability). For many cases, the design criteria provided are those included in the SRD.	This section updates the PSAR description of process systems. Process and instrumentation diagrams are provided for major systems. In addition, for those systems classified as Safety Design Class, the FSAR describes how the design requirements provided in the PSAR are reflected in the final design. For each system classified as Safety Design Class, the following are provided: 1 The specified safety function(s) with reference to Section 4.7 for the basis 2 The design basis 3 The design safety margins provided by the final design 4 Important quantitative design parameters met by the system design with their basis (e.g., heating, ventilation, and air-conditioning flow, and what established the minimum and maximum flow limits) 5 Material specifications 6 Established design limits and their basis (e.g., maximum pressure and temperature limits and what established these limits) 7 Instrumentation provided with attributes, including redundancy, diversity, in situ testability, environmental qualification, failure mode on loss of power, and the surveillance requirements as defined in section 4.8, "Controls for Prevention and Mitigation of Accidents". The means by which the monitoring requirements established in section 4.8 are also to be discussed in the FSAR. Potential adverse system interactions between systems of various design classification are addressed.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
4.7 Results of the Integrated Safety Analysis (ISA)	In addition to providing the results of the Process Hazards Analysis (PHA) and accident analysis, this section discusses the uncertainties of the PHA and accident analysis and relates these uncertainties to the required content of the construction authorization package. Section 4.7 provides the basis for the conclusion that resolution of the uncertainties will not have a significant impact on the construction authorization request. This discussion includes the following: 1 Characterization of the specific technical information that must be obtained to demonstrate acceptable resolution of the uncertainties 2 An outline and schedule of the program to resolve uncertainties 3 A discussion of the design and/or operational alternatives to resolve the uncertainties Section 4.7 of the PSAR also describes the preliminary Fire Hazard Analysis (FHA) and the consequence of each design-basis fire scenario, including the consequences in the area of origin and adjacent areas.	This section documents the resolution of any uncertainties identified in the PSAR. The FSAR describes the final FHA and all resolved uncertainties previously included in the PSAR and additional fire protection measures and equipment design.
4.8 Controls for Prevention and Mitigation of Accidents	Draft Technical Safety Requirements are included.	Final Technical Safety Requirements are included.
5.0 Radiation Safety	This chapter identifies the radiological exposure standards by which the radiation safety program is developed and the facility is operated to ensure the radiological safety of the public and workers. This chapter identifies the radiation protection criteria to be implemented in the facility design.	This chapter reflects the final facility design developed to the radiation protection criteria. It also describes the facility organization and plans for the conduct of operations. This chapter includes detail on facility operation within the radiological protection program exposure standards and other radiological protection requirements.
6.0 Criticality	The methodology for criticality analyses is provided to the extent the need to perform criticality calculation is found to be appropriate. The analyses may include margins in assumptions, bounding conditions, modeling and comparisons to the acceptance criterion, as appropriate, to account for uncertainties in the design and plans for operation.	Assumptions used in the PSAR to account for uncertainties in the design and plans for operations are removed from the FSAR criticality analysis to the extent that these uncertainties have been resolved. The FSAR describes the remaining criticality controls appropriate for the RPP-WTP.
7.0 Chemical Safety	The chapter identifies the program standards by which the chemical safety program is developed and operated to protect the public and workers against chemical hazards and hazardous situations. This chapter identifies criteria to be used for the development of chemical safety controls.	The chapter reflects the final facility design and facility organization and the developed plans for conduct of operations as related to chemical safety. This section also identifies the specific chemical safety controls to be implemented for protection of the public and workers.

Table G-2 Planned Differences Between Regulatory Guide 3.52 PSAR and FSAR Content [1]

Title	PSAR	FSAR
8.0 Fire Safety	This chapter describes automatic and manual fire protection features and administrative controls of the fire safety program. Also described are features of the ventilation system, building layout, and emergency egress routes important to fire safety.	Administrative controls to be implemented for the fire safety program are described, including final responsibilities of response forces, and the pre-fire plan used by firefighting personnel to suppress fires safely and effectively.
9.0 Emergency Management	This chapter identifies the applicable requirements and criteria to which the RPP-WTP Emergency Management Program are developed. A general outline of the program is presented and the relationship to the Hanford Site and local emergency management programs is discussed. Information is presented to demonstrate that the RPP-WTP staff will be able to attain an acceptable state of emergency preparedness by the time the facility becomes operational.	The FSAR discusses and references the specific emergency plan and implementing documentation prepared for the RPP-WTP. Specific aspects of all elements of the emergency preparedness program are discussed. Information is presented demonstrating the developed emergency preparedness program is compliant with applicable requirements, regulations, criteria, and guidance, and capable of responding to any operational emergency at the facility.
10.0 Environmental Protection	This chapter references the RPP-WTP Environmental Report submitted in Part A.	This chapter references the RPP-WTP Environmental Report as a new or revised Environmental Report and is not required to support the operating authorization request.
11.0 Deactivation and Decommissioning	This chapter identifies design considerations given to facilitate deactivation and decommissioning. It also discusses in general terms the planning, safety analysis, and regulatory considerations to be given to deactivation.	The chapter describes the specific design features included to facilitate deactivation and decommissioning. The level of detail for planning, safety analysis, and regulatory considerations to be given to deactivation is about the same as that provided in the PSAR. The FSAR is amended near the end of waste processing operation to provide more specific information regarding deactivation. (See Integrated Safety Management Plan Table 9-5.)

[1] *Standard Format and Content for the Health and Safety Sections of License Applications for Fuel Cycle Facilities*, Regulatory Guide 3.52, Revision 2, draft, US Nuclear Regulatory Commission, Washington, DC (NRC 1995).

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		**WTP New Location**	
Section	**Title**	**Section**	**Title**
	Introduction		Executive Summary
1.0	General Information	N/A	
1.1	Facility and Process Description		Executive Summary
1.1.1	Facility Description	2.3 2.4	Facility Overview Facility Structures
1.1.2	Process Description	2.5	Process Description
1.2	Institutional Information	N/A	
1.2.1	Identity and Address	1.1	Introduction
1.2.2	Activity	N/A	
1.2.3	Site Location	1.3.1	Geography

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
1.2.4	Type, Quantity, and Form of Licensed Material	3.3.2	Hazard Identification (Vol II - V)
1.3	Site Description	1.3	Site Description
1.3.1	Geography	1.3.1	Geography
1.3.2	Demography and Land Use	1.3.2	Demography and Land Use
1.3.3	Meteorology	1.4.1	Meteorology
1.3.4	Hydrology	1.4.2	Hydrology
1.3.5	Geology and Seismicity	1.4.3	Geology
2.0	Management Organization	17	Management, Organization, and Institutional Safety Provisions
2.1	Organization and Administration	17.3	Organizational Structures, Responsibilities, and Interfaces
2.1.1	Organizational Commitments, Relationships, Responsibilities, and Authorities	17.3	Organizational Structures, Responsibilities, and Interfaces
2.1.2	Management Controls	17.4	Safety Management Policies and Programs
2.2	Safety Committees	17.4.2	Safety Review and Performance Assessments
3.0	Conduct of Operations	11	Operational Safety
3.1	Configuration Management	17.4.3	Configuration Management
3.2	Maintenance	10.5	Maintenance
3.3	Quality Assurance	14	Quality Assurance
3.3.1	Management Commitment for QA Program	14	Quality Assurance
3.3.2	Scope of QA Program	14	Quality Assurance
3.3.3	Organizational Responsibility	14	Quality Assurance
3.3.4	QA Program Description	14	Quality Assurance
3.3.5	Graded QA Approach	14	Quality Assurance
3.3.6	Application of Graded QA to SSCs and Activities	14	Quality Assurance
3.4	Training and Qualification	12	Procedures and Training
3.4.1	Organization and Management of the Training System	12.4	Training Program
3.4.2	Trainee Selection	12.4	Training Program
3.4.3	Conduct of Needs/Job Analysis and Identification of Tasks	12.4	Training Program
3.4.4	Development of Learning Objectives as the Basis for Training	12.4	Training Program

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
3.4.5	Organization of Instruction Using Lesson Plans and Other Training Guides	12.4	Training Program
3.4.6	Evaluation of Trainee Mastery of Learning Objectives	12.4	Training Program
3.4.7	Conduct of On-The-Job Training	12.4	Training Program
3.4.8	Systematic Evaluation of Training Effectiveness	12.4	Training Program
3.5	Human Factors	13	Human Factors
3.5.1	Organization and Administration	13.3	Scope of Human Factors Process
3.5.2	Human Factors and Assessment of the Correction of Deficiencies	13.4	Human Factors Program
3.6	Audits and Assessments	17.4.2	Safety Review and Performance Assessment
3.7	Incident Investigations	13.4	Human Factors Program
3.8	Records Management	17.4.4	Document Control and Records Management
3.8.1	Organization and Administration	17.4.4	Document Control and Records Management
3.8.2	Types of Records	17.4.4	Document Control and Records Management
3.8.3	Record Handling Procedures	17.4.4	Document Control and Records Management
3.8.4	Record Storage and Protection	17.4.4	Document Control and Records Management
3.9	Procedures	12.3	Procedures Program
3.10	Testing Program and Preoperational Safety Review	10.3	Commissioning
3.11	Operational Practices	11.3	Conduct of Operations
4.0	Integrated Safety Analysis	3	Hazard and Accident Analysis
4.1	Site Description	1.3	Site Description
4.2	Facility Description	2.3 2.4	Facility Overview Facility Structures
4.3	Process Description	2.5	Process Description
4.4	Process Safety Information	3.3.3.X	Hazard Evaluation
4.5	Training and Qualifications of ISA Team	3.3.1	Identification of Work
4.6	ISA Methods	3.X	Hazard and Accident Analysis
4.7	Results of the Integrated Safety Assessment	3.3.3 (Vol II - V)	Development of Control Strategies

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
4.8	Controls for Prevention and Mitigation of Accidents	3.4 (Vol II - V)	Accident Analysis Methodology
4.9	Administrative Control of the ISA		Executive Summary
5.0	Radiation Safety	7	Radiation Protection
5.1	As Low As Reasonably Achievable (ALARA) Policy	7.1	Introduction
5.2	Organizational Relationships and Personnel Qualifications	7.1	Introduction
5.3	Radiological Safety Procedures and Radiological Work Permits (RWPs)	7.1	Introduction
5.4	Training	7.1	Introduction
5.5	Ventilation systems	2.6	Confinement Systems
5.6	Air Sampling	7.1	Introduction
5.7	Contamination Control	7.1	Introduction
5.8	External Exposure	7.1	Introduction
5.9	Internal Exposure	7.1	Introduction
5.10	Summing Internal and External Exposure	7.1	Introduction
5.11	Respiratory Protection	7.1	Introduction
5.12	Instrumentation	7.1	Introduction
5.13	Integrated Safety Analysis	3.4	Accident Analysis Methodology
5.14	Radioactive Waste Management	8	Hazardous Material Protection
6.0	Nuclear Criticality Safety	6	Criticality Safety Program
6.1	NCS Technical Practices	6.3 6.4	Criticality Limits and Concerns Criticality Controls
6.1.1	Process Analysis from the Integrated Safety Analysis	6.4.6	Application of Double Contingency Principle
6.1.2	NCS Evaluations	6.4.6	Application of Double Contingency Principle
6.1.3	NCS Limits	6.4.3	Administrative Controls
6.1.4	Validation and Use of Analytical Methods	6.4.4	Methodology for Determining Criticality Limits
6.1.5	NCS Control Methods	6.4.3	Administrative Controls
6.1.6	Criticality Accident Alarm System	6.6	Criticality Instrumentation
6.2	Administrative Practices	6.5	Criticality Protection Program
6.2.1	NCS Organizational Responsibilities	6.5	Criticality Protection Program
6.2.2	Configuration Management	17.4.3.2	Configuration Management Process
6.2.3	Maintenance	6.4.3	Administrative Controls

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
6.2.4	Quality Assurance (QA)	6.4.3	Administrative Controls
6.2.5	Training	6.5.4	Criticality Safety Training and Qualifications
6.2.6	Operational Inspections, Audits, Assessments, and Investigations	6.5.5	Criticality Safety Training and Qualifications
6.2.7	Written Operating Procedures	6.5.3	Administrative Controls
6.2.8	Materials Control for NCS	N/A	
6.2.9	Emergency Preparedness	6.6	Criticality Instrumentation
7.0	Chemical Safety	8	Hazardous Material Protection
7.1	Chemical Safety Responsibility	8.3	Hazardous Material Protection and Organization
7.2	Chemical Safety Approach	8.6	Hazardous Material Exposure Control
7.3	Chemical Safety Controls	8.6	Hazardous Material Exposure Control
7.4	Hazardous Waste Management	8.6	Hazardous Material Exposure Control
8.0	Fire Safety	N/A	
8.1	Organization and Conduct of Operations	18.3	Organization and Management
8.1.1	Organization and Management	18.3	Organization and Management
8.1.2	Training and Qualifications	18.4	Training and Qualifications
8.1.3	Fire Prevention Program	18.5	Fire Prevention Program
8.2	Fire Protection Features and Systems	18.6	Fire Protection Features and Systems
8.3	Manual Fire-Fighting Capability	18.7	Manual Fire-Fighting Capability
8.4	Fire Hazard Analysis	18.8	Fire Hazard Analysis
8.5	References	N/A	
9.0	Emergency Management	15	Emergency Preparedness
9.1	Description of On-Site and Off-Site Emergency Facilities	15.4.4	Emergency Facilities and Equipment
9.2	Types of Accidents	3.3.3	Development of Control Strategies
9.3	Classification of Accidents	3.3.3	Development of Control Strategies
9.4	Detection of Accidents	15.4	Emergency Preparedness Planning
9.5	Mitigation of Consequences	15.4	Emergency Preparedness Planning
9.6	Assessment of Releases	15.4	Emergency Preparedness Planning
9.7	Responsibilities of Licensee and Other Organizational Personnel	15.4	Emergency Preparedness Planning
9.8	Notification and Coordination	15.4	Emergency Preparedness Planning
9.9	Description of the Emergency Operational Center	15.4	Emergency Preparedness Planning

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
9.10	Information to be Communicated and the Parties to be Contacted	15.4	Emergency Preparedness Planning
9.11	Public Notification	15.4	Emergency Preparedness Planning
9.12	Training	15.4	Emergency Preparedness Planning
9.13	Procedures for Safe Shutdown and Recovery	15.3 15.4	Scope of Emergency Preparedness Program Emergency Preparedness Planning
9.14	Drills and Exercises	15.4	Emergency Preparedness Planning
9.15	Procedures for Identifying, Locating, and Controlling Hazardous Chemicals	15.3 15.4	Scope of Emergency Preparedness Program Emergency Preparedness Planning
9.16	Responsibilities for Developing and Maintaining Current the Emergency Program and Its Procedures	15.3 15.4	Scope of Emergency Preparedness Program Emergency Preparedness Planning
10.0	Environmental Protection	9.1	Introduction
10.1	Environmental Report	N/A	
10.1.1	Description of Proposed Action	N/A	
10.1.2	Purpose of Proposed Action	N/A	
10.1.3	Description of Affected Environment	N/A	
10.1.4	Discussion of Considerations	N/A	
10.1.5	Analysis of Environmental Effects of Proposed Action and Alternatives	N/A	
10.1.6	Federal and State Environmental Requirements	9.1.1 9.4	Permit Overview Radioactive and Hazardous Waste Processes
10.2	Environmental Safety Program	9.3	Radioactive and Hazardous Waste Management Program and Organization
10.2.1	Features for Contamination Control	2.3	Facility Overview
10.2.2	Environmental Monitoring Program	9.1 9.4	Introduction Radioactive and Hazardous Waste Processes
10.2.3	Emergency Plan	15.X	
10.2.4	Maintenance and Surveillance	9.3	Radioactive and Hazardous Waste Management Program and Organization
10.2.5	Configuration Management	17.4.3.2	Configuration Management Process
10.2.6	Organization and Management	9.3.1 17	Program Summary Management, Organization, and Institutional Safety Provisions
10.2.7	Quality Assurance	14.1	Introduction

Table G-3 Regulatory Guide 3.52 vs SAR Table of Contents Crosswalk

RG 3.52 Chapter		WTP New Location	
Section	**Title**	**Section**	**Title**
10.2.8	Training	9.3 9.7 (ERPP)	Radioactive and Hazardous Waste Management Program and Organization Environmental Radiological Protection Plan (ERPP)
10.2.9	Event Notification and Reporting	9.1.1	Permit Overview
10.2.10	Bibliography	9.8	References
11.0	Decommissioining	16	Deactivation and Decommissioning
11.1	Conceptual Decommissioning Plan	16.1	Introduction
11.1.1	Information for Conceptual Decommissioning Plan	16.4	Deactivation Requirements
11.1.2	Information for Total or Partial Cessation of Operations	16.5	Transition Readiness
11.1.3	Bibliography	16.7	References
11.1.4	Appendix A: Cost Estimating Tables	N/A	
11.2	Decommissioning Funding Plan and Financial Assurance Mechanisms	N/A	
11.2.1	Decommissioning Cost Estimate	N/A	
11.2.2	Financial Assurance Mechanism(s)	N/A	
11.2.3	Updating the Cost Estimate and Funding Level	N/A	
11.2.4	Bibliography	N/A	
11.2.5	Appendix A: Sample Sight Draft	N/A	
Appendix A	Radiation Protection Program Outline	7 RPP	Radiation Protection
Appendix B	Environmental Radiation Protection Program Outline	9.5 ERPP	Environmental Radiation Protection Plan (ERPP)

The SARs should include multiple volumes. Volume I should provide information that is applicable to more than one of the facilities (e.g., Pretreatment, Low-Activity Waste Vitrification, High-Level Waste Vitrification, and Balance of Facilities). Other volumes should be facility specific and contain, at a minimum, chapters 2, 3, 4, and 5.

Executive Summary

- **E.1 Facility Background and Mission**
- **E.2 Facility Overview**
- **E.3 Facility Hazard Classification**
- **E.4 Safety Analysis Overview**
- **E.5 Organization**
- **E.6 Safety Analysis Conclusions**
- **E.7 SAR Organization**
- **E.8 Summary of Significant Changes from the Preliminary Safety Analysis Report (FSAR stage)**

1 Site Characteristics

- **1.1 Introduction**
- **1.2 Requirements**
- **1.3 Site Description**
- **1.4 Environmental Description**
- **1.5 Natural Phenomena Hazards**
- **1.6 External Man-Made Threats**
- **1.7 Nearby Facilities**
- **1.8 References**

2 Facility Description

- **2.1 Introduction**
- **2.2 Requirements**
- **2.3 Facility Overview**
- **2.4 Facility Structures**
- **2.5 Process Description**
- **2.6 Confinement Systems**
- **2.7 Safety Support Systems**
- **2.8 Utility Distribution Systems**
- **2.9 Auxilliary Systems and Support Facilities**
- **2.10 References**

3 Hazard and Accident Analyses

3.1 Introduction

3.2 Requirements

3.3 Hazard Analysis Methodology

3.4 Accident Analysis Methodology

3.5 Hazard Classification

3.6 Common Cause and Common Mode Design Basis Events

3.7 Seismic Probabilistic Risk Assessment

3.8 Adherence to Risk Goals and Results

3.9 References

4 Important to Safety Structures, Systems, and Components

4.1 Introduction

4.2 Requirements

4.3 Safety Design Class Systems, Structures, and Components

4.4 Safety Design Significant Systems, Structures, and Components

4.5 References

5 Derivation of Technical Safety Requirements

5.1 Introduction

5.2 Requirements

5.3 Technical Safety Requirement Coverage

5.4 Derivation of Facility Modes

5.5 Technical Safety Requirement Derivation

5.6 Design Features

5.7 Interface with TSRs from Other Facilities

5.8 References

6 Criticality Safety Program

6.1 Introduction

6.2 Requirements

6.3 Criticality Limits and Concerns

6.4 Criticality Controls

6.5 Criticality Protection Program

6.6 Criticality Instrumentation

6.7 References

17 Management, Organization, and Institutional Safety Provisions

17.1 Introduction

17.2 Requirements

17.3 Organizational Structures, Responsibilities, and Interfaces

17.4 Safety Management Policies and Programs

17.5 References

18 Fire Protection

18.1 Introduction

18.2 Requirements

18.3 Organization and Management

18.4 Training and Qualification

18.5 Fire Prevention Program

18.6 Fire Protection Features and Systems

18.7 Manual Fire-Fighting Capability

18.8 Fire Hazard Analysis

18.9 Fire Protection for Filter Plenums

4.0 References

BNI, 2001. *Safety Requirements Document Volume II*, 24590-WTP-SRD-ESH-01-001-02. Bechtel National, Inc., Richland, Washington.

DOE-ORP, 2000. Contract No. DE-AC27-01RV14136, December 2000. US Department of Energy, Office of River Protection, Richland, Washington.

DOE-RL, 1998a. *DOE Regulatory Process for Radiological, Nuclear, and Process Safety for TWRS Privatization Contractors*, DOE/RL-96-0003, Revision 1. US Department of Energy, Richland Operations Office, Richland, Washington.

DOE-RL, 1998b. *Top-Level Radiological, Nucelar, and Process Safety Standards and Principles for TWRS Privatization Contractors*, DOE/RL-96-0006, Revision 1. US Department of Energy, Richland Operations Office, Richland, Washington.

NRC, 1994. *Integrated Safety Analysis Guidance Document*, draft NUREG-1513, distributed 8 August 1994, US Nuclear Regulatory Commission, Washington, DC.

NRC, 1995. Regulatory Guide 3.52, *Standard Format and Content for the Heatlh and Safety Sections of License Applications for Fuel Cycle Facilities*, draft Revision 2, US Nuclear Regulatory Commission, Washington, DC.